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SUBSONIC NUCLEAR AIRCRAFT STUDY

by Frank E. Rom

Lewis Research Center
Cleveland, Ohio

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION • WASHINGTON, D. C. • AUGUST 1968

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SUBSONIC NUCLEAR AIRCRAFT STUDY (U)

by Frank E. Rom

Lewis Research Center

SUMMARY

Unlimited aircraft flight duration or range offered by the use of nuclear energy is a unique capability that would have a major impact on transportation. An advanced technology program was initiated at the NASA Lewis Research Center in the summer of 1964 to assess the feasibility of safe, practical, publicly acceptable nuclear flight within the Earth's atmosphere. This report summarizes the status of this study as of May 1967. The main emphasis is placed on the safety measures necessary to gain public acceptance and on the technical feasibility of providing practical aircraft that has no more difficulties with regard to routine maintenance, handling, and all normal operations than conventional large aircraft.

To date there appear to be no fundamental or theoretical reasons why fission products, which could constitute a safety hazard if they escaped, cannot be contained during and after major aircraft accidents. Preliminary reactor fuel-element experiments with burnups of 10 percent indicate that aircraft reactor lives substantially greater than 1000 hours perhaps approaching 10 000 hours may be feasible. Preliminary conceptual design studies show that nuclear aircraft should be capable of carrying about 200 000 pounds (91 000 kg) of payload for an aircraft gross weight of 1 million pounds (454 000 kg) at Mach 0.8 and 36 000 feet (11 000 m).

Extensive analytical and experimental engineering studies are required to give assurance that proposed containment schemes and reactor lives approaching 10 000 hours can be achieved in practice.

Political aspects of nuclear aircraft safety are beyond the scope of this report.

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INTRODUCTION

The advantage of the unique capability of unlimited range that is offered by nuclear aircraft is readily apparent. It was this potential that originally motivated the now-cancelled Aircraft Nuclear Propulsion (ANP) program. A serious restriction that led to the cancellation of that program was that the gross weight was limited to about 500 000 pounds (227 000 kg). In addition, it was required that the aircraft carry enough chemical fuel to permit supersonic flight over the target. Serious compromises had to be made on the nuclear powerplant to stay within the mission-determined specifications. The severe gross-weight restriction excluded the possibility of complete shielding and long-life reactors, which resulted in impractical and unrealistic flight and ground operations. For example, an airplane crew would have received a lifetime radiation dose in 1 year; highly shielded radioactive hot shops would be required for simple repairs and normally routine inspection external to the reactor; and reactor overhaul every 100 hours was considered reasonable. Even the most rudimentary safety precautions for the protection of the general population in the event of an accident could not be made. Within the last 5 years, it has become apparent that several projected Air Force missions require long range, long endurance, and/or large payload capacity. Nuclear aircraft, if developed, would seem to be well suited for providing world wide mobility independent of remote or foreign bases. If the nuclear aircraft is to be developed, it must become publicly acceptable and operationally sound. For the nuclear aircraft to gain public acceptance the potential hazard to the general population must be no more than that acceptable for chemical aircraft today. The radiation dose received by flight crews, ground crews, passengers, and the general population during normal operation or accidents should be limited to no more than that allowed by the Federal Radiation Council. The nuclear aircraft should have a reasonably long reactor life between overhauls (minimum of 1000 hr, preferably approaching 10 000 hr) so that elaborate hot-shop work on radioactive components would be minimal. In general, to be of interest, the nuclear aircraft should have no more complex operational and maintenance procedures than conventional large chemically powered aircraft. This report summarizes the objectives, the status, and the progress of a study being conducted at Lewis to determine whether safe, practical, and publicly acceptable nuclear aircraft are feasible.

NASA ATMOSPHERIC NUCLEAR TRANSPORT STUDY

In August of 1964, Lewis initiated a low-level program, complemented with a small out-of-house contract effort, entitled Atmospheric Nuclear Transport Study. The study is being conducted to assess the technical prospects for development of realistic, practical,

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safe, publicly acceptable nuclear aircraft. It is not arbitrarily limited by specific missions inasmuch as uses not thought of at present could well be the major applications of the future. Nuclear aircraft should be thought of in terms of the increases in time global mobility and transportation capability that it could give rather than as fulfilling the requirements of a specific mission.

Goals

The goals of the project are as follows:

(1) Determination of the technical feasibility of practical, safe, maintainable, subsonic nuclear transport aircraft

(2) Definition of the major problems requiring research or development

(3) Development and/or demonstration of technology as required to permit feasibility assessment

Ground Rules

The following ground rules were established at the beginning of the program in order to fulfill the requirements that the aircraft be practical, safe, and maintainable:

(1) The approach should be conservative and practical, using technology that has been demonstrated by laboratory tests or experience.

(2) The time between reactor overhaul should approach 10 000 hours.

(3) Utmost attention must be paid to safety.

(4) Operations and maintenance should be at least equivalent to those for chemical aircraft.

Approach

After the goals and ground rules were established, the following approach was formulated:

(1) Study three competing candidate propulsion systems simultaneously

(2) Perform state-of-the-art surveys of each

(3) Make detailed conceptual design studies of each

(4) Perform laboratory-scale experimental and analytical studies to verify or demonstrate the state-of-the-art technology where required

(5) Update integrated conceptual propulsion-system designs on a continuing basis as new information is obtained

(6) Perform integrated aircraft and powerplant optimization studies to fix design variables for the propulsion-system conceptual design studies

(7) Produce practical, safe, conceptual propulsion-system designs which use demonstrated technology and are fully integrated with the appropriate aircraft

Study Areas

The work on this study has been divided into four areas. The remainder of this report presents the work in these areas in the following order:

(1) Safety

(2) Long-life powerplant components

(3) Integrated propulsion-systems studies

(4) Integrated aircraft optimization studies

Primary emphasis has been placed on safety because we think this is by far the largest single obstacle to acceptance of nuclear aircraft.

Long life has been singled out as the second study area. During the ANP program nuclear powerplants with lifetimes of the order of 100 hours were considered as a reasonable goal. Such short life seems to be unacceptable for a practical airplane. Accordingly strong emphasis in this study has been placed on evaluating the feasibility of powerplants with lives approaching 10 000 hours.

Integrated propulsion-system studies which attempt to combine long-life reactors with unit shielding and with safety provisions such as containment vessels and crash-energy-absorbing structures have never been attempted before. Implementing each of these provisions without regard to mutual interaction would probably yield rather unwieldy powerplants. Designing systems which take advantage of multiple- or dual-function components can greatly reduce the penalties of each requirement alone. For example, the reactor shield requires large amounts of material for attenuating radiation. This material should be incoporated within the reactor-shield assembly so that it performs other functions such as reactor-core structure, containment-vessel material, and perhaps also as crash-energy-absorbing structure.

Finally, integrated aircraft optimization studies are being made to balance reactor-shield and powerplant-operating variables against aircraft structure and aerodynamic variables. Studies, such as this, guide the powerplant designer in selection of such parameters as reactor density, ducting size, heat-exchanger operating conditions, fan bypass flow ratio, fan pressure ratios, and compressor pressure ratios. In addition, continually updated optimization studies provide a measure of the overall performance capability anticipated for nuclear aircraft.

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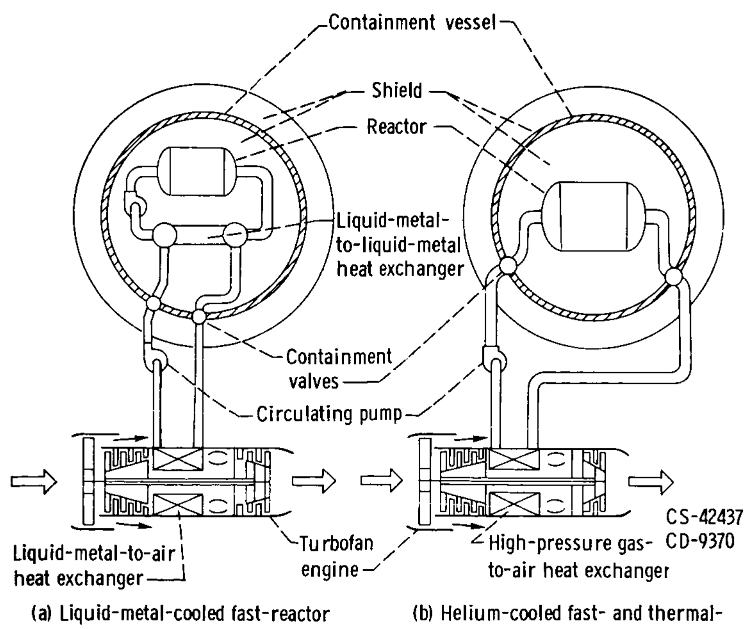
Systems Under Investigation

It was decided to investigate at least three basically different reactor and coolant combinations for aircraft nuclear powerplants that are representative of the most promising concepts. These three concepts are being thoroughly investigated in parallel. The concepts were limited to closed systems to the exclusion of open systems because of the requirement that no fission products be dumped into the atmosphere. In addition, open systems require large reactors because of the poor heat-transfer characteristics of air. The resultant weight of shielding required for open systems is prohibitive.

The reactor-coolant combination represents the most basic difference between various nuclear propulsion systems. It was therefore decided to use the same type of propulsive thrust unit for each system and to concentrate comparisons on the reactor assembly. The turbofan engine was chosen as the thrust unit for all systems. Comparative performance would then reflect only the effect of the reactor-coolant system.

The systems under investigation are shown in figure 1. The fast-reactor liquid-metal system which was one of the two favored systems in the ANP program is shown on the left side of figure 1. Liquid metal is heated in a fast-neutron spectrum reactor, circulated through a liquid-metal-to-liquid-metal heat exchanger, and then pumped through the reactor to complete the primary circuit. The liquid metal in the secondary circuit is heated in the liquid-metal-to-liquid-metal heat exchanger. The secondary liquid metal is then circulated through the liquid-metal-to-air heat exchanger of a tur-



(a) Liquid-metal-cooled fast-reactor system.

(b) Helium-cooled fast- and thermal-reactor systems.

Figure 1. - Nuclear propulsion systems for aircraft.

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bofan engine. Circulation is accomplished by means of the pump on the cold leg of the liquid-metal-to-air heat exchanger.

Because the primary liquid metal becomes radioactive in its transport through the reactor, the primary loop is completely contained within the containment vessel and within the shield. A containment vessel is provided so that in an emergency situation all fission products and activated materials are contained. Emergency quick-acting containment valves are provided in the coolant ducts that lead through the containment vessel. These valves must be designed to provide a positive seal against leakage of radioactive material. Reactor shielding is located both within and without the containment vessel in a minimum-weight arrangement.

The second system in this study is the fast-spectrum-reactor inert-gas system shown in figure 1(b). Helium, is heated in the reactor and circulated to the high-pressure gas-to-air heat exchanger of a ducted-fan engine. From this heat exchanger the helium is pumped back through the reactor. Helium is not used as working fluid; it merely serves to transfer heat from the reactor to the heat exchanger of the propulsive system. A secondary heat exchanger is not used in this system because helium does not become activated as it passes through the reactor. There is a possibility in this system that radioactivity can be transferred out of the containment vessel in the event of a reactor fission-product release. As in the case of the liquid-metal system, a containment vessel with containment valves in the coolant lines is shown. Radiation shielding is provided both within and without the containment vessel.

The third system (fig. 1(b)), which is basically the same as the second system, uses a thermal-neutron spectrum reactor instead of a fast-neutron spectrum reactor. Helium is the heat-transfer fluid as in the second system. The thermal reactor uses a water moderator. Since thermal reactors are, in general, larger than fast reactors, the normal assumption might be that a fast system would be better because the reactor would be smaller and, therefore, the shield would be smaller and lighter. In real systems there are many other factors which must be considered. For example, ducting, valves, control systems, auxiliary systems, pumps, and other components which must be shielded may require more volume than the reactor itself. In light of practical factors such as these, it is not possible to say beforehand which reactor type is best.

It might be concluded that a liquid-metal system is superior to a gas system because of the better heat-transfer performance that might be expected of liquid-metal systems. (Actually, helium at pressures of 1000 to 2000 psi (6.9×10^6 to 13.8×10^6 N/m^2) is as good a heat-transfer fluid as liquid metals in terms of heat-removal capacity per unit of reactor-core volume). Better heat-transfer performance would lead to a smaller reactor and possibly a lighter shield. Many other factors could cancel the benefits of the possibly higher heat-transfer capacity of liquid metals. One such factor is apparent from figure 1. All liquid metals become radioactive to a degree which requires shielding and containment in the event of an accident. The liquid-metal-to-liquid-metal heat exchangers and primary pumps must therefore be housed within the containment vessel. This fact, together with the requirement for many auxiliary components within the shield, causes the reactor to be only a fraction of the volume that must be shielded. The use of the extra heat exchanger within the shield virtually eliminates the problems that might occur due to release of fission products into the external circulating system. On the other hand, liquid metals introduce difficult corrosion and mass-transfer problems. Considerations of this nature make it apparent that a particular system can not be chosen as superior to another without careful and thorough study. Such study must include fairly detailed conceptual design layouts.

In summary, both fast and thermal reactors and both liquid-metal and inert-gas cooling systems are being considered in this study. Each of the three systems, the fast liquid-metal, the fast inert-gas, and the thermal inert-gas systems, are receiving thorough and complete treatment of comparable depth to permit a fair and consistent comparison.

SAFETY

The single greatest obstacle to acceptance of nuclear aircraft is the problem of safety. Accordingly, the primary emphasis in this study is in this area. The aircraft should be designed so that in normal operation or in the worst accident no person, whether he be a passenger or a member of the flight crew, the ground crew, or the general population, should receive a greater radiation dose than is permissible by the Federal Radiation Council guides. Our aim is that the nuclear-powered aircraft be no more hazardous than conventional, large, chemically powered aircraft. This aim represents a strong departure from the ANP philosophy. The only provision for general population safety made at that time was that the aircraft fly in well-defined corridors that would minimize the number of people that could be exposed in the event of a serious accident. Because of the severe weight limitation, the reactor shield was divided. Some shielding was located around the reactor and some around the crew. The dose levels outside the crew compartment were extremely high, in most cases about 10 000 to 1 million times that allowed by normal standards. The ground-crew dose was so high that routine ground operations had to be carried out by men who would dash in to do their maintenance tasks and dash out again with their weekly quota of radiation exposure used up in a fraction of a week's work.

The following discussion shows that the concept of complete reactor shielding which gives the flight crew, the ground crew, and the passengers doses within acceptable limits is feasible. It may also be feasible, in principle at least, to design an aircraft reactor that would not release fission products in the worst possible accident, including a 90° impact on granite at 600 feet (183 m) per second. By feasible in principle we mean that there are no fundamental or theoretical reasons why it could not be done.

The safety studies are divided into two catergories, safety in normal operations and safety in emergency situations. Safety in normal operations provides chiefly for safety to the flight crew, the ground crew, and the passengers. Also of interest in normal operation is the prevention of accidents. Safety in emergency situations provides chiefly for safety to the general population in the event of major or minor aircraft accidents. Each of these areas is discussed in the following sections.

Safety in Normal Operations

Normal operation safety considers unit shielding, flight procedures, and fail-safe and redundant systems. Unit shielding is concerned with minimizing the exposure of personnel to radiation from the core. Flight procedures are concerned with avoiding the possibility of an accident which could release fission products by limiting the flight conditions during which nuclear power can be used. Fail-safe and redundant systems are concerned with providing the sensing and detecting equipment (1) to detect any equipment malfunctions, (2) to determine the aircraft altitude, attitude, position, and proximity to solid objects be it surface or mountains, and (3) to provide for safe reactor control and operation with a high degree of reliability. Techniques must be worked out for redundant and fail-safe operation of all equipment. It is not anticipated that any new techniques, instruments, or electronics need be invented for this purpose. All the required functions have already been performed or demonstrated in space and defense systems. Further discussion of fail-safe and redundant equipment is not necessary. The normal operation safety problems of unit shielding and flight procedures are discussed in the following sections.

Unit shielding. - As stated previously, a practical nuclear airplane requires unit shielding to shield the flight crew, the ground crew, and the passengers from the radiations leaving the nuclear reactor. By unit shielding is meant complete shielding around the reactor so that the flight crew, the ground crew, and the passengers will be free to carry out all operations without any concern for the radiation from the reactor. Complete shielding can only be accomplished at the expense of high shield weight. The problem for a practical nuclear airplane is therefore one of providing adequate shielding with a minimum of weight.

Table I lists the allowable radiation doses used in this study. The doses to anyone in any way connected with the aircraft including the general population were limited to values generally used in the atomic energy industry. They are shown in table I. The total dose any

[redacted]

TABLE I. - RADIATION DOSES

	Minimum distance from reactor		Exposure time per year, hr	Allowable dose, rem	Dose[a] received, rem
	ft	m			
Flight crew	130	40	1000	5.0	2.5
Ground crew	10	3	1000	5.0	2.5
Passengers	40	12.2	100	.50	.25
Cargo	20	6.1	100	10	10
Individual dose	---	----	per accident	25	-----
General population	---	----	per accident	10	-----

[a]Does not take into account a factor of 10 or so reduction in dose rate due to extra shielding provided by aircraft structure, equipment, cargo, or shield shaping, except in the case of passenger dose where a factor of 10 reduction in dose is allowed.

TABLE II. - SHIELDING CODES

Code	Use
LEPRECHAUNS (UNC), one-dimensional optimization code	Selection of shield materials, optimization of layers
SANE-2 (UNC), one-dimensional Monte Carlo neutron code	Parametric studies
SAGE (UNC), one-dimensional Monte Carlo gamma code	Parametric studies
UNC-SAM-2 (UNC), three-dimensional Monte Carlo code	Shielding for conceptual designs
O5R (ORNL), three-dimensional Monte Carlo code	Duct penetrations

crew member may receive is 5 rem per year. This dose is known as the total occupational dose. Passengers may receive no more than 0.5 rem per year. Individual dose in an accident may not exceed 25 rem. General population doses may not exceed 10 rem per accident. Doses beyond 10 rems are grounds for legal claims by the recipients. In the case of the flight crew, the dose rate at the crew compartment, which is assumed to be 130 feet (40 m) from the reactor is $2\frac{1}{2}$ mrem per hour, which gives a total dose for the year of $2\frac{1}{2}$ rems or one-half the allowable dose. The ground crew is assumed to re-

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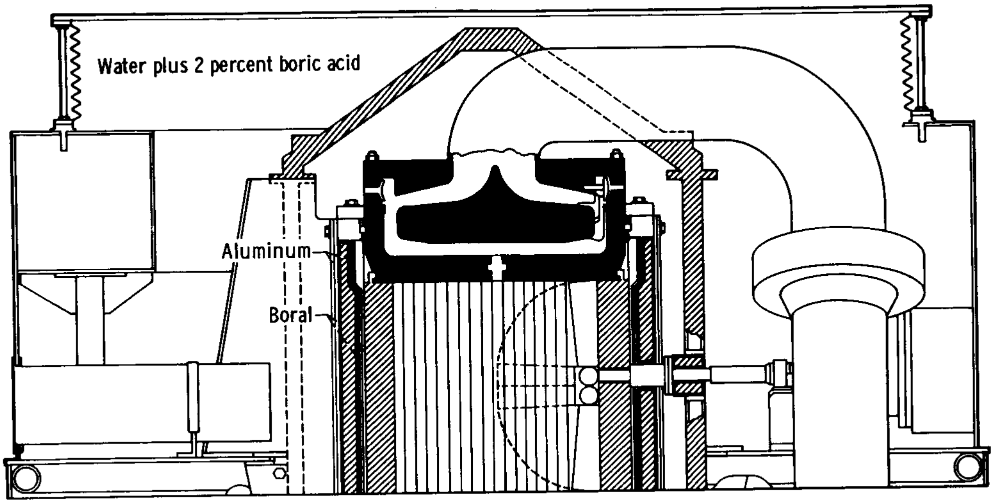
ceive the same dose as the crew, also in 1000 hours but at 10 feet (3 m) from the reactor. The full-power dose to cargo at 20 feet (6.1 m) was 50 mrem per hour which in 100 hours would lead to 5 rems. The passengers for commercial application were limited to a dose of 0.25 rem per year and were assumed to be no closer than 40 feet (12.2 m) from the reactor center. It was assumed that aircraft structure, equipment, cargo, luggage, or slight shield shaping reduces the dose in the passenger compartment by a factor of 10. If credit is not taken for shielding provided in this manner, the additional shield weight required for passengers would be less than 5000 pounds (2270 kg) which is beyond the accuracy of the calculation. In the event of an accident, the doses to the general population are limited by definition to less than 10 rems.

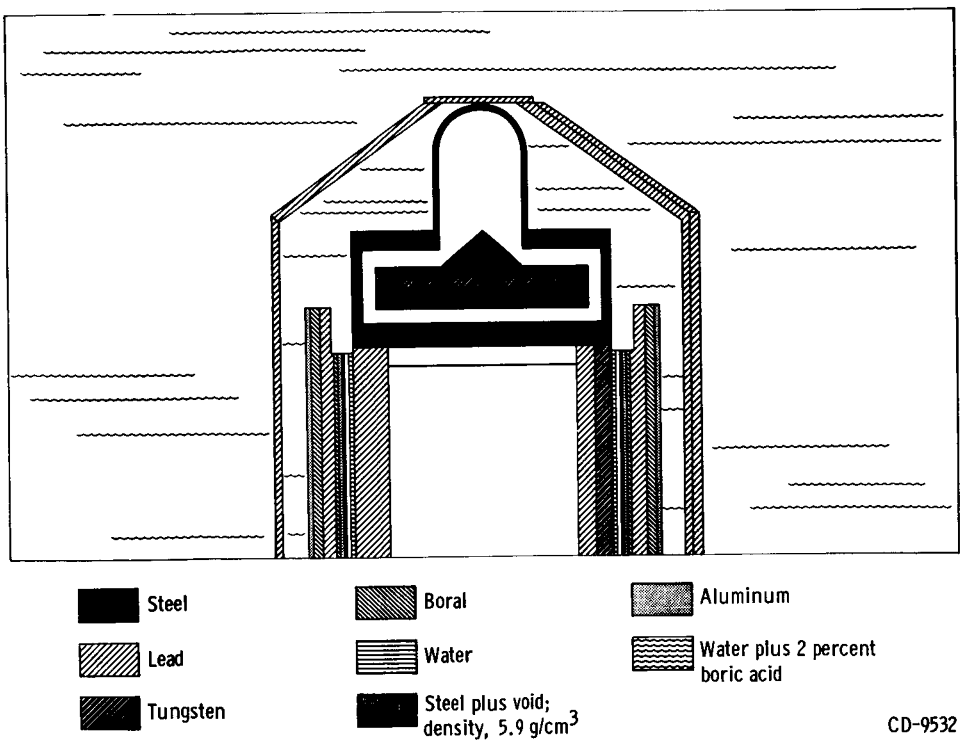
Because the shield is the most significant weight in the powerplant, and because safety of the crew, both flight and ground, the cargo, and the general population is involved, a survey was made to determine the best and most reliable shielding computer programs available. A great deal of progress has been made since the days of the ANP program in calculating and optimizing shields. Several of the latest shielding programs, listed in table II, were used for a variety of reasons.

LEPRECHAUNS, developed by the United Nuclear Corporation, is a one-dimensional computer program which permits the selection of shield materials and layer thicknesses. It considers any variety of shield materials, and selects or rejects them on the basis of minimum shield weight. In so doing, the best thickness of each of the layers of materials is also determined. The emphasis in LEPRECHAUNS is on the selection of shield materials and layer thicknesses. A simplified shield calculation technique is used in this computer program in order to save calculation time. When shield materials and layer thicknesses have been selected, SANE-2 and SAGE (ref. 1) are used. These one-dimensional Monte Carlo shielding computer programs for neutron and gamma attenuation are used for parametric shield studies. The parametric studies are used in our aircraft and powerplant optimization programs to determine proper engine and aircraft operating variables. Another United Nuclear Corporation computer program UNC-SAM2 (ref. 2) is used for special detailed calculations. This three-dimensional Monte Carlo program is quite sophisticated and has a longer running time than the other programs used. It permits calculation of complicated three-dimensional shielding-reactor arrangements defined by our conceptual design studies. An Oak Ridge computer program (O5R, ref. 3) which is also a three-dimensional Monte Carlo program is used for shield duct penetration studies. Shield duct penetration is of concern particularly for gas-cooled systems where ducts which must pass through the shield could provide possible paths for radiation leakage. Experimentally determined doses from a real reactor-shield configuration were compared with a calculation of precisely the same configuration.

The experimental check made of the three-dimensional UNC-SAM2 shield program



(a) Cross section of actual ML-1 reactor-shield assembly.



(b) Computer representation of ML-1 reactor-shield assembly.

Figure 2. - Experimental check of UNC-SAM-2 computer program with ML-1.

is illustrated in figure 2. The cross section of the ML-1 reactor (ref. 4) and its shield assembly is shown in figure 2(a). In figure 2(b) is shown the representation that was used in calculating the dose with the UNC-SAM2 computer. The calculated value of the total gamma and neutron dose at 500 feet (152 m) from the reactor core differed from the measured dose by less than 20 percent. Other checks have been made on these shield programs with equally good results for simpler configurations. For the purpose of this study, the shield calculation techniques available are entirely satisfactory.

The results are shown in figure 3 of a parametric study made of the shielding of a spherical reactor with a power density of $3\frac{1}{2}$ megawatts per cubic foot which is in the range of interest for long-lived reactors. The shield weight is plotted as a function of reactor power in megawatts. Five different shield configurations were considered: (1) conventional lead and water shields, (2) tungsten and water shields, (3) depleted uranium and water shields, (4) uranium hydride, titanium hydride, and water shields, and (5) uranium hydride, titanium hydride, and lithium hydride shields. Each of these shields was first optimized with the LEPRECHAUNS computer program or its equivalent. SANE and SAGE calculations were then used to obtain the curves shown in figure 3.

The upper three shield weight curves (fig. 3) represent more or less conventional shields that were considered during the ANP program. The only difference between these shields and those from the ANP program is that they have been better optimized with regard to shielding layer dimensions. The lowest curve, which uses the best shield materials as selected by the LEPRECHAUNS code, indicates a possible reduction in shielding weight. For example, the best hydride yields a reduction of about 35 percent

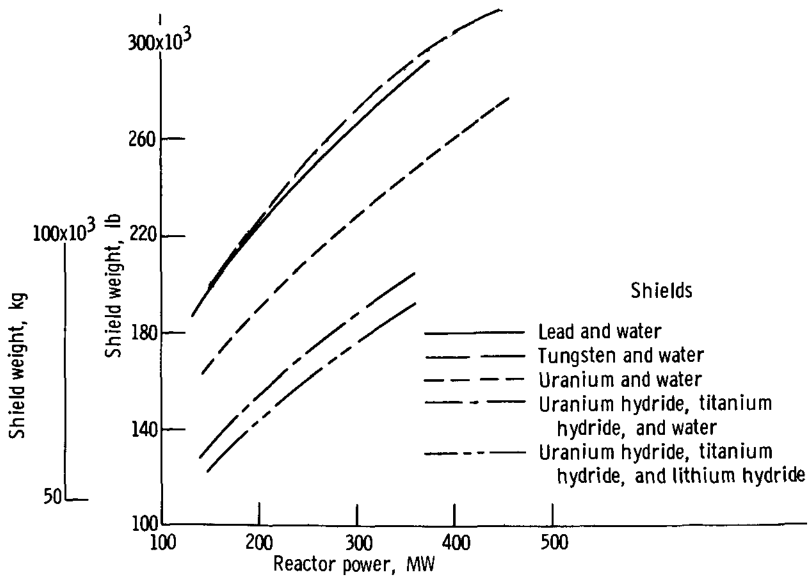


Figure 3. - Parametric shield study. Shielded volume power density, 3.5 megawatts per cubic foot (123 MW/m^3).

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in weight over an optimized lead-water shield. At 250 megawatts this represents an 85 000-pound (46 000-kg) reduction in weight. A 30 000-pound (13 600-kg) weight saving results from using uranium-water shields rather than lead-water shields. There is little difference between tungsten-water or lead-water shields. Whether hydride or water is used for the outer shield also has little effect on shielding weight. The biggest savings occurs from the use of heavy metal hydrides. The difficulty with the heavy metal hydrides is that there is little experience with them, and they are difficult to fabricate and become thermally unstable at temperatures above a few hundred degrees Fahrenheit.

Because hydride shielding requires a substantial increase in the state-of-the-art, this study is based on the use of uranium-water shields. In the 200 to 250 megawatt range, which is of interest for 1-million-pound (454 000-kg) airplanes, the shield weights will be of the order of 200 000 pounds (91 000 kg).

Flight procedures. - The techniques considered as methods for providing safety through flight procedures are as follows: The nuclear airplane should take off under chemical power with the reactor "shutdown" in a condition ready for the worst possible accident. The aircraft should climb chemically to a safe altitude before the reactor is started. A safe altitude is that altitude which assures sufficient time to shutdown the reactor before any possible normal or accidental physical contact with any object or terrain. Reactor shutdown is defined as reducing the reactor fission-generated power to zero, switching the turbofan engines to chemical operation, placing the reactor into a normal afterheat-removal condition, and generally placing all reactor and shield cooling and auxiliary systems into a configuration or condition ready to accept the worst possible crash or accident. The time to accomplish shutdown is expected to be of the order of tens of seconds. Enough chemical fuel reserve is provided for about 1 hour of flight without reactor power. Finally descent and landing by chemical power is specified below the safe altitude. The aim of these procedures is to prevent any possible normal or accidental contact with an object or the ground while the reactor is operating under power and not ready for a crash. With these procedures, there will always be sufficient time (probably of the order of 20 sec or so) to prepare the reactor system for a major accident should it occur unexpectedly.

To further understand the significance of flight procedure safety, consider a nuclear airplane flying at a safe altitude. The reactor is designed so its normal shutdown configuration is adequate to handle the maximum credible accident without the release of fission products. If it should take n seconds, for example, to put the reactor into the normal shutdown configuration, an n-second warning is required before any possible contact with any solid object or the ground. The safe altitude then is defined as that altitude which provides sufficient time so that, no matter what happened to the airplane,

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the reactor could not reach the ground in less than n seconds, the time necessary to place the reactor into an accident-ready situation. In the case of mountain peaks or other aircraft, the reactor must again be shutdown, so that impact with either of these must not be made in less than the n seconds. Redundant and fail-safe contact-prediction equipment is needed to assure this shutdown. The chance of determining that a possible contact could occur with n seconds should be virtually 100 percent, considering the advanced state of technology of guidance and control systems used in the space and defense programs.

Safety in Emergency Situations

The four areas of concern for emergency-situation safety are (1) nuclear hazards analysis, to determine just what is of concern, (2) containment during impact, (i.e., not rupturing the containment vessel around the core), (3) containment after impact, (i. e., the prevention of containment-vessel rupture following a reactor-core meltdown), (4) fuel-pin leak in a one-loop system, where the possible radiation dose should fission products leak to the heat exchangers and ducting in the aircraft is of concern.

Nuclear hazards analysis. - A great deal can be done to minimize the probability of a major aircraft accident, but it is not reasonable to say that one will never occur. If no special provisions are made, fission products will almost certainly be released in major accidents. A hazards analysis was made by assuming that virtually all the fission products were released at the point of aircraft impact on land in order to see what the effect of this release would be.

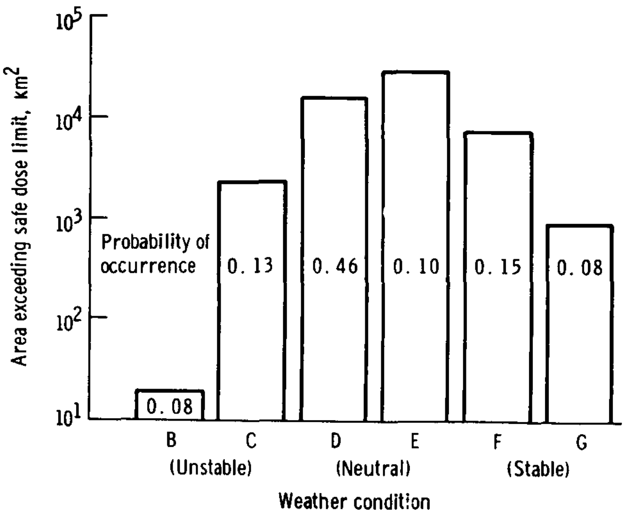


Figure 4. - Land release of fission products following 1000 hours at 250 megawatts. Full fission-product release; average area exceeding safe dose limit, 11 900 square kilometers (4600 sq mi).

TABLE III. - AVERAGE WORLD WEATHER CONDITIONS[a]

Weather condition	Frequency of occurrence, percent	Wind velocity, km/hr
A (most unstable)	0.01	4.3
B	.07	8.1
C	.13	11.8
D (neutral)	.46	16.8
E	.10	11.5
F	.15	6.1
G (most stable)	.08	3.0

[a]Taken at 23 worldwide weather stations.

[redacted]

In this study average world-wide weather conditions were determined since this is a major factor in spreading fission products which could cause damage to population and agriculture. Altogether 23 world-wide stations were used to determine the average conditions shown in table III. With the use of this weather data, calculations were made to determine the extent of area affected by radioactivity (see fig. 4). For this calculation, it was assumed that virtually all the fission products were released instantaneously from a reactor that had previously been operated at a power of 250 megawatts for a period of 1000 hours. In the worst case the general population within an area of 30 000 square kilometers downwind of the crash could receive a dose greater than 10 rems. Weighting the area affected for each weather condition with its probability of occurrence gives an average area affected of about 12 000 square kilometers or about 4600 square miles. We feel that such an event would be unacceptable.

The dollar costs of the damages caused by such accidents were also investigated and found to be low compared to operating costs even though the cost per accident would be high. The assumption was made in this calculation that the probability of a major nuclear aircraft accident is equal to that for current, large, chemical jet aircraft. It is likely that the probability of a major accident for a nuclear aircraft would be lower than that for current chemical aircraft because of the extra safety measures provided the nuclear aircraft. Although the costs of accidents were not serious, the possibility of personal injury to the general population beyond that encountered with chemical aircraft has led to the following conclusion. We feel that nuclear aircraft will not be acceptable for flight over populated areas unless it can be demonstrated that virtually no fission products will escape from the reactor in the most serious crash situation. Therefore, the major goal of the atmospheric Nuclear Transport study is to determine the feasibility of providing nuclear power for aircraft with no greater hazard to the general population than for conventional, large, chemically powered aircraft.

Impact occurring on water has also been studied. It was assumed that all fission products were released under water following 1000 hours operation at 250 megawatts. In such an accident, only noble (inert) fission gases escape to the atmosphere. All other fission products will condense or be dissolved in the water. The major results are as follows: Whole body submersion is no concern after 1 day (i.e., a person could swim in the water directly in the spot where the reactor impacted after 1 day). There would be some local seafood contamination that would cause some economic loss in a localized region for less than 2 months. The airborne dose from the noble (inert) fission gases that would be released would be less than the allowable dose at a distance 5 miles downwind from the impact point. Although the contamination problem over water is much less than over land, provisions for fission product containment may be required for overwater flights as well as for overland flights for political or psychological reasons.

The effect of a nuclear excursion on fission-product release has not been evaluated. The only nuclear excursion known to be safe at this time is one that would take place under water. Until such time as the feasibility of containing excursions is analyzed, the systems must be designed so that no nuclear excursion will take place in the worst accident. This requirement can be met in thermal reactors, and there is a possibility that this may be true for fast reactors also.

The results of our nuclear hazards analysis can be summarized as follows. First, acceptably low probability of nuclear excursion by engineered safeguards in aircraft reactor systems is required. Secondly, no fission-product containment is required for unrestricted overwater flight beyond 5 miles of shore. Third, fission-product containment is required for unrestricted overland flight.

Fission-product containment. - Fission-product containment in the event of a major aircraft accident is the most challenging problem facing acceptance of nuclear aircraft. Containment problems can be divided into two categories. The first, impact survival, concerns the prevention of the rupture of the containment vessel on impact, which might occur at velocities as high as 600 feet (183 m) per second. The second, post-impact survival, concerns the prevention of a melt-through of the containment vessel due to reactor afterheat generation.

The worst possible foreseeable accident is the aircraft with its nuclear powerplant impacting terrain at high velocity. The aircraft, along with the containment vessel and the reactor contained within it, has an enormous amount of kinetic energy when traveling at high speed. This kinetic energy must be absorbed by external means and the containment vessel decelerated uniformly and as slowly as possible, in order to prevent the containment vessel from bursting open on impact.

For the purpose of this study, the rather severe case of near-normal impact at 600 feet (183 m) per second on an unyielding surface, such as granite, was assumed to determine whether it was possible to conceive of any system that would survive without excessive penalties. The question might be asked, "What happens after such an impact if the reactor assembly cartwheels, tumbles, rolls, or bounces down a mountainside, across a field, hitting boulders, buildings, and other obstacles?" The impact velocity involved in these secondary situations is lower than 600 feet (183 m) per second. The kinetic energy to be absorbed is therefore much lower because kinetic energy varies directly as the square of the velocity. The impacts in these cases can occur on any side of the reactor assembly at random; therefore, all sides must be protected. Considering both these factors results in energy absorber weights that are small compared with the 600-foot (183-m) per second near-normal impact system.

Another question that might be asked is, "What happens if the flight trajectory is not coincident with the axis of the energy absorber, so that the high-velocity absorber is not directly in front on impact?" The high-velocity-impact absorber can probably be designed to take care of impacts that are up to 30° off-axis. It is doubtful that high velocity could occur for greater angles than this, but this point might require more study. At high velocities, the aircraft would decelerate rapidly and break up before angles of 30° could be obtained (due to aerodynamic forces). Whether the remnants of the aircraft with the reactor assembly could maintain speeds of 600 feet (183 m) per second needs further study. If the probability that this could occur is significant, parachutes or other deceleration devices could be deployed to slow down and/or orient the reactor assembly to any desired condition prior to impact.

The possibility of fire after a crash is also of concern. However, because a nuclear aircraft would carry relatively little fuel and because it could be placed in the outer wing panels at a considerable distance from the reactor (where it can be used to relieve wing loads) the effect of fire is expected to be minimal. The reactor assembly is massive and would be designed to take temperatures in the vicinity of 1700° to 1800° F (1200° to 1260° K) to take care of core meltdowns and afterheat removal. The duration of a chemical fire would be short compared with the duration of afterheat sources. The total energy absorbed by the reactor assembly from a fire would probably be small compared with the afterheat energy that the reactor assembly is designed to handle.

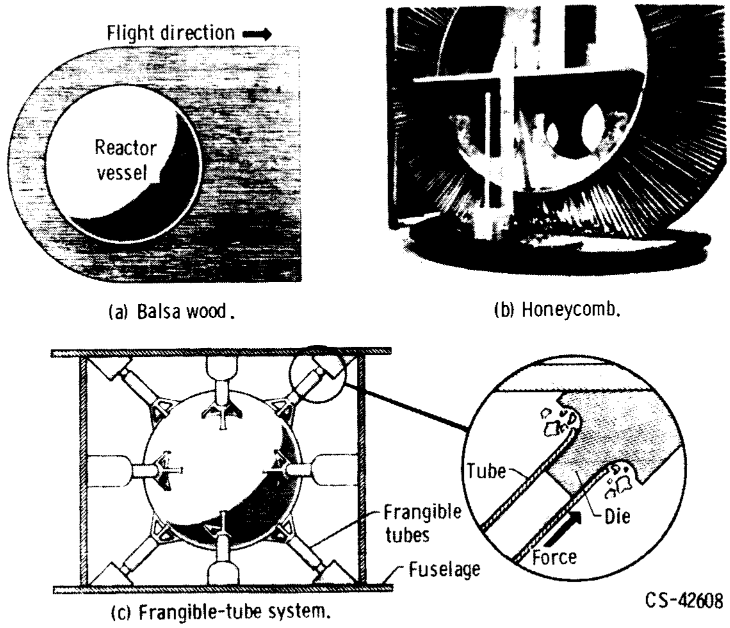
Detailed study of all phases of nuclear aircraft accidents is beyond the scope of this report. Safety problems are treated only in sufficient depth to determine whether or not it is possible in principle to prevent containment system rupture in major aircraft accidents.

Some schemes that have been considered for absorbing the kinetic energy are shown in figure 5. Balsa wood has often been proposed as a good energy absorber. Figure 5 shows how balsa could be utilized to absorb the kinetic energy of an aircraft reactor vessel. A greater thickness of balsa is provided in the forward direction since the highest impact velocity would occur from this direction.

Crushable honeycomb made of thin sheets of metal, plastic, or other materials can be used in the same fashion as balsa wood. An aluminum honeycomb designed to fit around a cylindrical object is also shown in figure 5. The crushed portion of the honeycomb indicates the part that has absorbed energy in this test. The amount of energy that can be absorbed per pound of honeycomb material is chiefly a function of the strength to density ratio of the material, once an optimum honeycomb design has been determined.

The frangible-tube technique (ref. 5) (fig. 5) utilizes tubes which are extruded over a mandrel by the deceleration forces occurring during impact. Severe working and subsequent breaking up or tearing of the tube as it is extruded is the mechanism fcr absorbing the energy. The amount of energy that can be absorbed by this system is determined by the strength and ductility of the tube material and by the design of the mandrel, which should maximize the work put into the tube without failing itself.

As shown in figure 6, balsa wood is able to absorb about 20 000 foot-pounds of energy



(a) Balsa wood.

(b) Honeycomb.

(c) Frangible-tube system.

Figure 5. - Passive impact-energy absorbers.

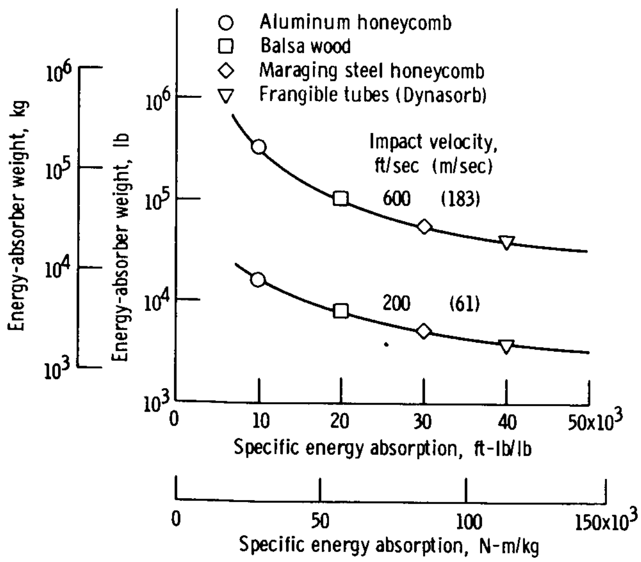


Figure 6. - Impact-energy absorption. Reactor - shield-assembly weight, 250 000 pounds (113 000 kg).

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per pound (6100 (m-kg)/kg). If the reactor-shield - containment-vessel assembly weighs 250 000 pounds (113 000 kg), it would take about 100 000 pounds (45 000 kg) of balsa wood to absorb the energy of this package, if it impacted at 600 feet (183 m) per second from the forward direction and 250 feet (76 m) per second from all other directions. The next technique discussed in figure 5 was to use a honeycomb material as an energy absorber. The crushing of the honeycomb can absorb from 10 000 to 30 000 foot-pounds per pound (3050 to 9150 (m-kg)/kg) depending on the strength and properties of the material of which the honeycomb is made. Aluminum honeycomb absorbs about 10 000 foot-pounds per pound (3050 (m-kg); maraging steel honeycomb absorbs about 30 000 foot-pounds per pound (9150 (m-kg)/kg). The weight of the maraging-steel-honeycomb energy-absorbing system would be about 55 000 pounds (25 000 kg) compared with 100 000 pounds (45 400 kg) for the balsa wood system. The third system discussed makes use of frangible tubes to absorb energy. This idea was developed by Langley Research Center (ref. 5). A variation called Dynasorb has been developed by the Lockheed Corporation. These units have been measured to absorb 50 000 foot-pounds per pound (15 200 (m-kg)/kg) of material. For an impact velocity of 600 feet (183 m) per second, a frangible-tube impact-absorption system weighing about 40 000 pounds (18 200 kg) would absorb the kinetic energy of the 250 000-pound (113 000-kg) reactor-shield - containment-vessel assembly. The overall weight penalty may be less inasmuch as the frangible-tube units could be made part of the aircraft structure.

The foregoing techniques for absorbing kinetic energy are termed "passive" systems because they do not require warning, triggering, deployment, or activation of any kind prior to an accident. These systems are always ready to perform their function. Other systems, termed "active", involve operations such as deployment of flaps, parachutes, firing of retrorockets, and activation of the other deceleration devices prior to impact. Active systems have been considered but are not discussed in this report. It is deemed sufficient to show that it is feasible in principle by some readily visualized schemes to absorb all the kinetic energy of the reactor-shield assembly at impact without a prohibitive weight penalty. Feasibility in principle does not mean demonstrated in practice. A great deal of design and experimental work is necessary to demonstrate feasibility in practice.

The decelerations experienced during impact with the energy-absorbing systems will be as high as 300 g's. The containment vessel must be designed to withstand both the external and internal forces that will act upon it during such decelerations. Quick-closing valves must be provided to seal off all containment-vessel penetrations during and after impact. The loads that are applied during impact last for about 50 milliseconds and are steady-state rather than shock-type loads because of the relatively long duration of the decelerating forces. This fact, of course, greatly simplifies design and test procedures.

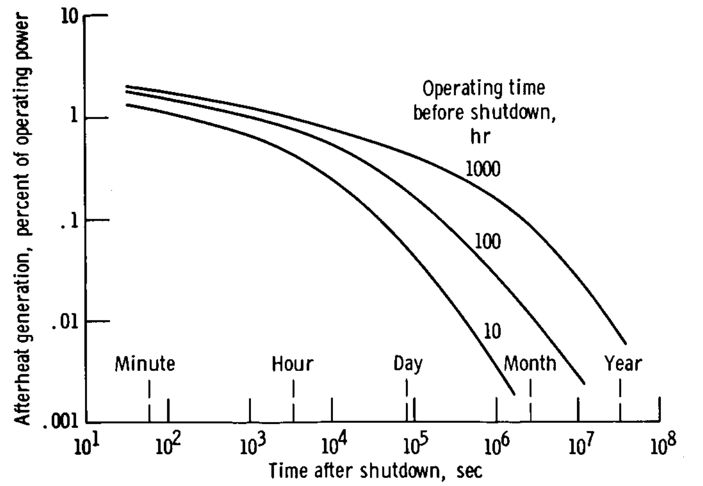

Figure 7. - Afterheat generation.

Following impact, the containment vessel must survive the effects of reactor meltdown which will very likely occur. Probably no aftercooling system external to the containment vessel could survive the most severe impacts. Preliminary calculations were made of the conditions to be expected due to reactor meltdown. The heat generated as a function of time after reactor shutdown is shown in figure 7. Approximately 1 hour after shutdown, the afterheat power is of the order of 1 percent of the power of the reactor before shutdown. Several days are needed to reduce the afterheat power to 1/10 of a percent of the reactor power.

The containment-vessel surface temperature required to remove the afterheat, if the containment vessel were exposed to air after an accident, is shown in figure 8. The afterheat would be removed by a combination of radiation and thermal convection from the surfaces of the containment vessel, assuming that the containment-vessel surface is exposed directly to the air. In the case of the balsa and honeycomb techniques, provisions must be designed into the system to permit the balsa or honeycomb to burn off or melt away, exposing the containment shell to the air. The system using frangible tubes can be envisioned as a cage of frangible tubes surrounding and supporting the containment vessel while maintaining a specified minimum clearance from the ground.

In figure 8, the reactor is assumed to have operated for a period of 1000 hours at a level of 250 megawatts before shutdown or impact. The surface temperature for an assumed uniform heat distribution is plotted as a function of time in minutes after emergency shutdown. Curves are shown for three different containment-vessel diameters: 8, 10, and 12 feet (2.4, 3, and 3.7 m). In the case of the 10-foot (3-m) containment vessel, at about an hour after shutdown the surface temperature reaches a maximum of 1700° F (1200° K) for a short period time. The total time above 1600° F (1140° K) is of the order of 1 hour. The total time above 1200° F (920° K) is about 10 hours. Contain-

[redacted]

ment vessels fabricated from high-strength oxidation-resistant allov would be adequate, if they were no smaller than 10 feet (3 m) in diameter.

If the containment vessel were allowed to ontact the surface of the Earth, the insulating qualities of the soil would allow the temperature of portions of the containment vessel to exceed the temperatures shown in figure 8. The temperatures, in this case, could reach about 2000° to 3000° F (1360° to 1920° K), or the approximate melting point of the materials of the Earth's surface. If it is necessary to provide for containment in this situation, the containment vessel must be fabricated of a refractory metal which is

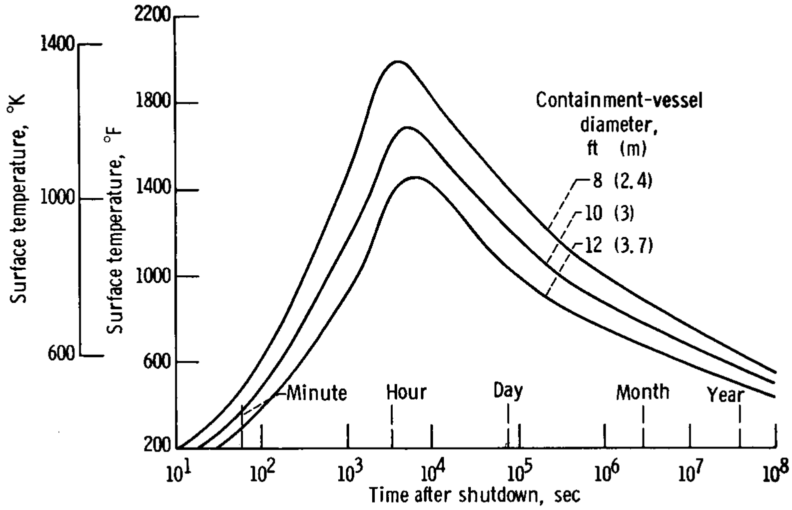

Figure 8. - Post-impact containment-vessel surface temperature following 1000 hours at 250 megawatts.

ductile enough to survive distortions which occur during impact. Tantalum or tantalum-alloys could be suitable for this application. Since refractroy metals are not oxdiation resistant, coating materials which are oxidation resistant at high temperatures and ductile at low temperatures must be provided. The problem of refractory-metal containment vessels can be avoided by designing the containment vessel to avoid Earth contact (i. e., by surrounding it with a suitable shock-absorbing frame, as discussed previously in this section).

The peak surface temperature of the containment vessel if it is submerged in water is shown in figure 9. The peak surface temperature is plotted as a function of containment-vessel diameter. From this consideration alone, the containment vessel could be as small as 5 feet (1.5 m) in diameter before the surface temperature starts to increase rapidly because of film boiling.

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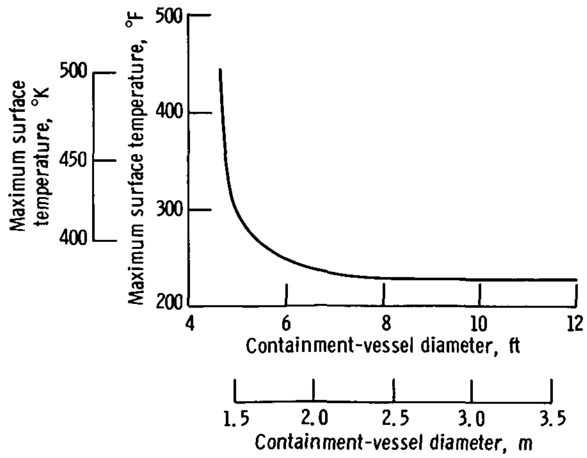

Figure 9. - Post-impact water submersion following 1000 hours at 250 megawatts.

TABLE IV. - FUEL-PIN LEAK IN ONE-LOOP SYSTEMS

Reactor power, MW	250
Reactor operating time, hr	>100
Total fission-product inventory, curies (dis/sec)	10^9 (3.1×10^{19})
Number of fuel-pin segments in reactor	60 000
Releasable activity per pin segment, curies (dis/sec)	6×10^3 (2×10^{14})
Crew dose rate per ruptured pin segment, rem/hr	0.024
Allowable number of ruptured pin segments (10-hr exposure for total of 25 rem)	100

The final safety consideration examined is the case of a fuel-pin leak in a one-loop system. Table IV presents data for this situation for a reactor power of 250 megawatts and a reactor operating time prior to fission-product release of greater than 100 hours. No significant increase in fission-product activity occurs beyond 100 hours. These data apply, therefore, for all preimpact operating times greater than 100 hours. The total fission-product activity built up in the core is about 10^9 curies (3.1×10^{19} dis/sec).

A typical reactor design using fuel pins would have about 60 000 fuel-pin segments, each isolated from one another in a single reactor. The largest amount of activity that could be released from each pin segment is about 6000 curies (2×10^{14} dis/sec). (The actual release most probably will be less than this for a variety of reasons.) If this activity is released into the ducting and heat exchangers of a one-loop system, the dose rate that the crew would receive for each ruptured pin segment would be about 0.024 rem per hour. For a 10-hour exposure time, which should be more than adequate to bring the aircraft to its home station, 100 fuel-pin segments could be ruptured at once. The

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crew would then receive the maximum allowable dose level of 25 rems as set by the Federal Radiation Council.

The decay of activity with time after the fission products have been released has been ignored. After 10 hours the activity of the released gases would be reduced considerably. (An allowance for shielding due to the duct walls, the heat exchanger, and the engine and aircraft structure materials has been made.) Considering past experience with fuel pins (ref. 6), the possibility of one fuel-pin leak should be virtually zero. The probability of 100 fuel-pin leaks occurring simultaneously should be extremely remote. Considering this probability and the relatively low dose from 100 leaks if they do occur leads to the conclusion that fuel-pin leakage in one-loop systems should not be a major problem.

In the discussion of safety, it has been assumed that no nuclear excursions would occur. In-flight excursions are excluded by virtue of fail-safe design techniques. The question of excursions on impact may be of greater concern. In the case of thermal reactors, nuclear excursions can be excluded by prior removal of the moderator material. For example, if water is used as the moderator, the water can be removed from the reactor whenever the aircraft is within a specified time, such as 20 seconds, of any possible contact with another solid object. (This procedure would be part of the normal safety and shutdown routine.) In the case of fast reactors, the core would have to be designed in such a way that collapse of the core would not cause criticality. Another possibility for fast reactors is to prevent core collapse caused by the loads that occur during impact. Excursions that might take place during post-impact meltdown could be prevented by providing poison diluent materials that would mix with the molten core materials. Water flooding would not be a factor inasmuch as the containment vessel is designed not to rupture during any impact. Water could not, therefore, enter the reactor core.

Safety Studies Outlook

Heavy emphasis in this study is placed on safety. The philosophy of this approach is that no one, whether he be a passenger or the flight crew, the ground crew, or the general population, should receive any dose beyond any that would be recommended by the Federal Radiation Council limits. This philosophy results in large powerplant weights which can only be accommodated by considering larger aircraft than heretofore have been considered for nuclear propulsion. The outlook for incorporating these features at present is summarized in the following paragraphs.

Normal operations can be made safe by designing unit shielding to protect the flight crew, the cargo, and the ground crew. Flight procedures can be arranged so that the reactor-shield assembly is automatically placed in a normal shutdown mode of operation before there is any possibility of an accident. The reactor is automatically shutdown and engines switched to chemical power whenever it is possible, in normal or abnormal conditions, to contact terrain or any solid object within a certain specified time period. The time specified should be greater than that required to place the reactor in the normal shutdown mode of operation. The normal shutdown condition is designed to accept all accidents, including the maximum accident that can be postulated, without releasing fission products.

Fail-safe detection equipment with a high degree of redundancy and reliability, which warns that it is possible to make contact with terrain or any solid object within the specified time limit, is a key requirement for safe nuclear flight. With the experience gained in our national space and military programs, the development of such equipment should be of a merely routine rather than a research nature.

In the case of emergency situations, our preliminary studies have shown that fission-product containment for land impact is feasible in principle (i. e., there is no fundamental reason why it cannot be done). A thorough engineering study to prove this feasibility is presently being conducted. Fission-product containment after land impact (during the period of reactor afterheat generation) seems to be feasible in principle, providing the containment vessel is made large enough, approximately 10 feet (3 m) in diameter. There are two possible post-impact conditions. The first is that the containment vessel is air-cooled, which means that after impact it must not be in contact with the ground. The containment-vessel temperature will be low enough in this case to permit the use of ordinary oxidation-resistant pressure vessel materials. The second possibility is that the containment vessel is in contact with the ground after impact. The containment-vessel temperatures might be high enough in this case to require the use of refractory materials such as tantalum alloys. Protection with oxidation-resistant cladding materials that are ductile at low temperature to withstand the impact would be necessary in this case.

It seems reasonable to conceive of an energy-absorbing-system design that would prevent the reactor-shield assembly from contacting the ground and thus permit air cooling and the use of conventional pressure vessel materials. The reactor-shield assembly in this scheme would be supported in the center of a cage-like structure made of frangible tubes that would permit impact from any direction. Studies have been initiated to explore the engineering reasonableness of this approach.

Although the contamination problem for overwater flight accidents is much less severe than for overland accidents, fission product containment may be required for political or psychological reasons.

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LONG-LIFE POWERPLANT COMPONENTS

In this section is discussed the possibility of achieving long operating life, approaching 10 000 hours, between major overhauls. The air-handling portions of the propulsion system can achieve such operating times, as demonstrated by the commercial jet engines of today. The chief problem is providing a reactor with long life without increasing its size to extremes which would cause large shield-weight penalties. In addition, the air heat exchanger must also be designed for similar life without excessive weight penalties. Long-life reactors and heat exchangers imply that routine maintenance between major overhauls is identical to that for chemical aircraft. No special facilities would be required at each airport to handle radioactive components. The only requirement for radioactive handling would be at the major overhaul base. For reactor lifetimes of 10 000 hours, the aircraft would have to fly to the major overhaul base only once every 2 or 3 years.

The two general long-life problem areas, the reactor and the air heat exchanger, are discussed in this section. The reactor long-life problems are further subdivided into (1) fuel-element problems, (2) reactor-structure problems, and (3) reactor neutronics problems. Fuel elements represent the primary area of concern for long-life reactors because they contain the fissionable material and therefore are subject to the problems caused by high fuel burnup. Reactor-structure problems concern the effects of prolonged nuclear radiation on the properties of reactor-core structure and support materials. Reactor neutronics problems are chiefly concerned with the reliability of techniques used for designing reactors that make efficient use of the fissionable material contained within the core.

The requirement of relatively long reactor life in this study indicates a markedly different approach to practical nuclear aircraft from that of the ANP program. We feel that for a nuclear aircraft to be practical and economically feasible, the reactor section of the powerplant should have a lifetime between major overhauls comparable to that for the propulsion systems of current aircraft, that is, approaching 10 000 hours.

The second general problem area of concern for long-life powerplants is heat exchangers. Both the liquid-metal and the inert-gas systems presently require heavy heat exchangers. These must be designed to be thermodynamically efficient and reliable, while minimizing weight. The properties of the tube and header materials which comprise the major portions of the heat exchanger must be known, to provide minimum weight together with reliability. These properties must be determined in the environment in which the heat exchanger operates. In heat exchangers that heat air, oxidation resistance is of great concern. High-temperature creep strengths are of primary concern for tubes and header materials. Fabrication becomes of great concern for these heat exchangers because of the large number of tubes or large amount of heat-transfer

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surface required, with the attendant large number of welded joints. These joints must be leak-tight to prevent escape of liquid-metal or high-pressure-gas coolants. The heat exchangers must be compact so that they can be placed within the engine envelopes and not create undue drag or structural-weight penalties.

Discussion of long-life components, such as pumps, valves, ducting seals, expansion joints, and other items necessary for a complete powerplant, is omitted inasmuch as these problems are not as severe as those of the reactor and the heat exchangers. If nuclear aircraft are to be developed, these components will most certainly require more attention.

Long-Life Reactors

Few long-life reactors have been built that operate at temperature levels above 1200° to 1300° F (920° to 980° K), which are required for reasonable nuclear aircraft performance. Although technology exists for such powerplants, demonstration at operating conditions of particular interest to nuclear aircraft is lacking. In this section the problems, the state-of-the-art, and the programs being conducted at Lewis for demonstrating technical feasibility of long-life reactors are discussed, as well as long-life reactor fuel, long-life reactor structures, and long-life reactor neutronics.

<u>Long-life reactor fuel.</u> - The primary requirements of long-life fuel for aircraft application are (1) fission of relatively high fractions of the fissionable material (high burnup) and (2) relatively high operating temperatures compared to conventional nuclear powerplants. The burnup desired is at least 5 percent, but preferably 10 percent or higher. The operating temperatures of interest range from a minimum of 1500° F (1090° K) to about 2500° F (1640° K). In this section are described the basic types of fuel-element materials and designs that can possibly achieve these requirements. Typical data obtained from all available reactor programs are shown to give an indication of the progress to date. The data obtained in the Lewis state-of-the-art technology-verification experiment programs are also presented herein.

Two basic ways in which the reactor fuel is incorporated within reactor materials are shown in figure 10. One way is to disperse the fissionable material, such as uranium dioxide, in a metal matrix. The metal matrix provides support for the fuel particles, and in addition, provides for efficient heat removal from the fuel particles through the high conductivity of the matrix metal. Cermet-type fuels (fig. 10(a)) are usually constructed to provide an additional measure of containment by cladding the surfaces of the matrix.

Another typical fuel form is the bulk fuel pin. A pin made of a suitable high-temperature material that is compatible with the reactor coolants is completely or par-

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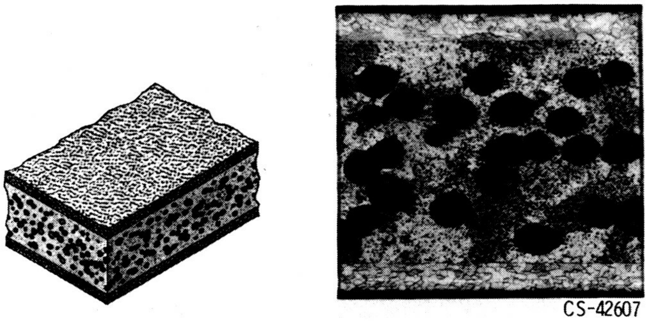
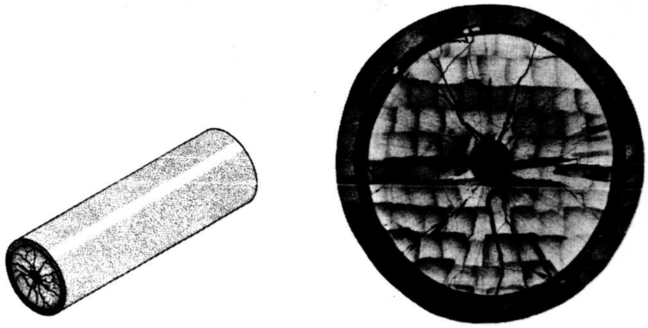


(a) Cermet.



(b) Pin.

Figure 10. - Fuel-element types.

tially filled with bulk uranium dioxide. The central regions of the UO_2 in fuel pins may operate in the molten state because of the relatively low thermal conductivity of uranium dioxide. The cross section in figure 10(b) shows a bulk UO_2 fuel pin that has operated with the center molten. The cladding in this fuel-pin arrangement serves as a container for the gaseous fission products that are formed as the uranium fissions. The greater the number of fissions, the greater will be the fission gas generation, the higher will be the internal gas pressure, and therefore the greater will be the required tube wall thickness.

In the case of the cermet material, the fission gases released must be contained in the limited amount of voids within the UO_2 or the metal matrix. The UO_2 particles within the matrix can be made porous to provide for larger amounts of fission. The metal matrix can also be made porous so that more fission gases can be contained before the plate fails.

In both fuel forms, high-strength materials that can operate at 1500° to 2500° F (1090° to 1640° K) are required. The materials must also be compatible with the reactor coolant and the coolant system.

Now these basic fuel forms can be formed into fuel-element designs suitable for use

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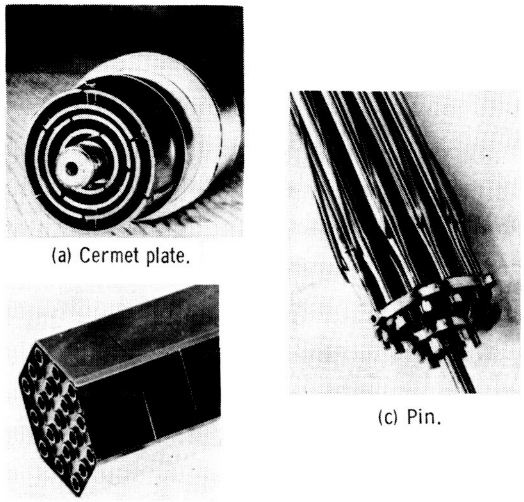

(a) Cermet plate.

(b) Cermet matrix.

(c) Pin.

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Figure 11. - Fuel-element designs.

in a reactor is shown in figure 11. The basic materials are formed into shapes or designs that are satisfactory from heat-transfer, pressure-drop, and structural points of view. The cermet fuel in the form of plates can be rolled into curved shapes to form a concentric ring-type fuel element. The reactor coolant flows between the rings where it is heated to desirable operating temperatures. Cermets can be formed into a matrix-type element, such as the hexagonal element shown in figure 11(b). In this case, the metal UO_2 matrix is pierced with heat-transfer passages to provide for heating of the reactor coolant. The passages are clad to prevent loss of fuel and fission products from the matrix. The element is also clad on the outside and end faces to prevent losses from these surfaces.

The most common form of fuel-element design utilizes fuel pins, as shown in figure 11(c). Fuel pins are clustered together to form a suitable heat-transfer and structural geometry. End fixtures support the tubes in the longitudinal and radial directions. Positive spacing is provided by spiral wires which wrap each individual pin. The spiral wires also aid in heat transfer. The reactor is merely an assembly of a number of elements such as these sufficient to provide the necessary surface area for heat removal and also to provide sufficient fuel for long life and to maintain a nuclear chain reaction.

Many reactor programs (civilian, military, and space) are being conducted in this country and the world that can provide experienced knowledge. In this study, we have been and are continuing to keep abreast of all fuel-element research for which the data is accessible. The approach has been to consult directly with the source of the data whenever possible. A sampling of data collected for cermet fuels is given in table V. High burnups have been recorded at temperatures to 1550° F (1120° K) in stainless-

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TABLE V. - CERMET FUEL DATA

Fuel material	Source temperature		Burnup, percent of UO_2	Source
	^{o}F	^{o}K		
UO_2 - W	3200	2000	0.2	Argonne National Laboratory
UO_2 - stainless steel	1125	880	55	Battelle Memorial Institute
	1250	950	16	
	1400	1030	12	
UO_2 - Nb	1800	1260	2.0	Battelle Memorial Institute
UO_2 - stainless steel	1550	1120	14	Battelle Memorial Institute (GCRE)
	1730	1220	6	
UO_2 - Mo	4200	2600	.2	General Electric (Hanford)
UO_2 - stainless steel	750	670	35	General Electric (KAPL)
	1275	960	18	
UO_2 - W	3000	1920	1.0	General Electric (NMPO)
UO_2 - W - 10 percent Re	2900	1870	~1.0	
UO_2 - stainless steel	1470	1070	4	Nuclear Development Associates
UO_2 - stainless steel	650	620	5	Pratt and Whitney Aircraft
	750	670	4	
	800	700	3	
UO_2 - Nb	1800	1260	.2	Pratt and Whitney Aircraft

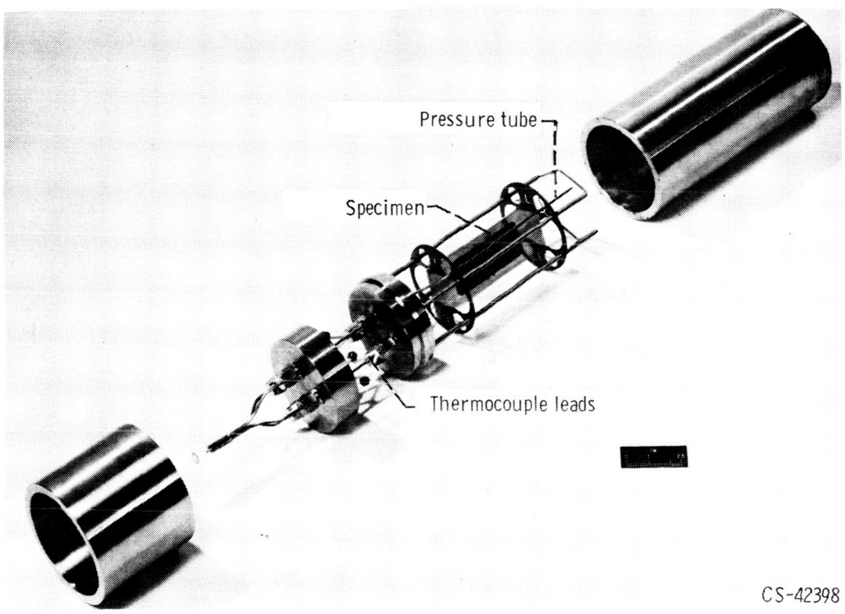

Figure 12. - In-pile irradiation capsule.

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steel - uranium dioxide cermets. The burnup data at higher temperatures are generally no higher than the 2 percent obtained with a niobium cermet operating at 1800° F (1260° K). Tungsten-UO_2 and tungsten-rhenium-UO_2 cermets have achieved burnups of the order of 1 percent at temperatures to 3000° F (1920° K). It is apparent that for temperatures higher than 1500° F (1090° K) no data exist for fuel burnups of 5 percent or greater that appear to be necessary for a practical long-life aircraft.

It is possible to manufacture many cermet fuel forms, but they have not been evaluated at the conditions of interest. Therefore, sample fuel plates were purchased from many sources, which represented their best experience and technique. Some cermets were also built at Lewis based on our assessment of the experience in the field. These sample plates were then irradiated in a reactor to determine how much burnup these state-of-the-art molybdenum cermet fuel plates would take. Figure 12 shows an exploded view of one of our in-pile capsule tests. The sample fuel plate, about 1 by 2 inches (2.5 by 5 cm), clad on all sides and edges, is mounted in a frame that supports the plate vertically along the centerline of the capsule. The plate is cooled by thermal convection through a high-pressure inert gas, helium. The capsule is pressurized to 2500 psi (17.2×10^6 N/m^2) by means of the tube shown. Gas is continuously bled from the capsule to detect the occurrence of a fission-product leak. The temperature of the plate was measured by means of thermocouples mounted along the unfueled edges.

Typical results from these cermet in-pile irradiations are shown in table VI. Early results showed burnups of 1 percent or less caused by faulty plates or difficulties in running the experiments. Later test results obtained with 85-percent-dense UO_2 show burnups in the range of 4 to 10 percent at an operating temperature 2700° F (1750° K) which is 200° or 300° F (100° or 200° K) in excess of that required at the hot spot of

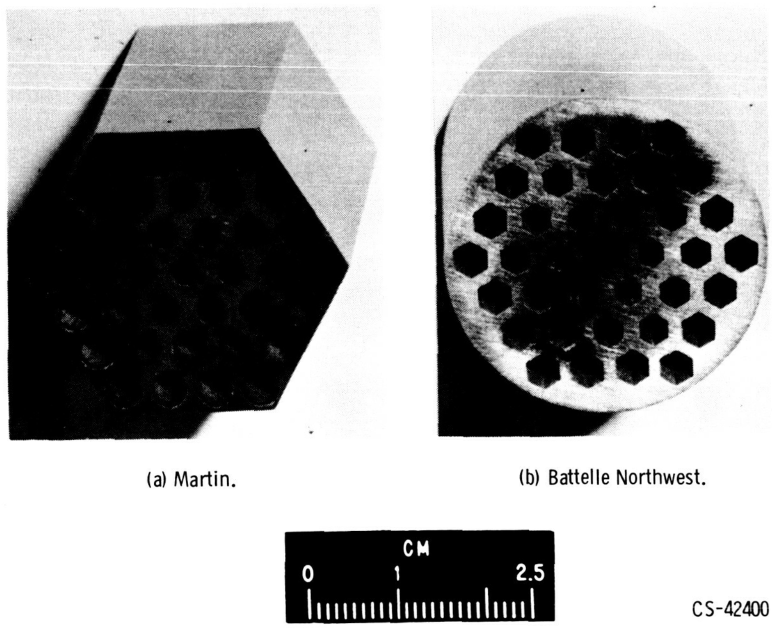
TABLE VI. - LEWIS MOLYBDENUM-PLATE

IRRADIATIONS

Cladding process	UO_2 density, percent	Operating temperature		Burnup, percent of UO_2
		°F	°K	
Roll clad	0.98+	2500	1650	~0.1
Roll clad	.98+	2500	1650	~1
Sintered	~.85	2300	1530	.92
Roll clad	.98+	2300	1530	3.82
Roll clad	~.85	2700	1750	4.47
Vapor cementation	~.85	2700	1750	~7
Roll clad	~.85	2700	1750	~6
Roll clad	~.85	2700	1750	~10

reactors for aircraft application. The most successful plates were those that were roll clad or vapor cemented. One capsule was still running as of May 1964 and at that time the specimen had a burnup in excess of 10 percent without releasing fission products. It would appear reasonable, for this study, to assume a maximum burnup of 5 percent for cermet-type fuels operating at temperatures as high as 2700° F (1750° K). However, it is worth exploring burnups into the 10-percent range or higher, inasmuch as this would offer the possibility of more than doubling the core lifetime.

To demonstrate that cermet-type fuel can be fabricated into reasonable fuel-element geometries, several contractors, were asked to furnish, on a best-efforts basis, cermet fuel stages of specified composition and heat-transfer geometry. Two contractors have successfully fabricated hexagonal-type honeycomb fuel elements, which are shown in figure 13. It was specified that the heat-transfer passages in the center of the cermet



(a) Martin. (b) Battelle Northwest.

Figure 13. - Hexagonal-type honeycomb cermet fuel stages.

fuel stages be smaller than those near the outer edges to provide a desirable fuel distribution and thus achieve maximum burnup. The Martin specimen has circular flow passages; for the Battelle Northwest specimen, hexagonal passages were specified and led to a honeycomb cell structure with variable web thickness, which is desirable from a heat-transfer standpoint. Both stages were fabricated as the vendor predicted.

A vast amount of information about pin-type fuel materials has been obtained as a result of a survey of world-wide power reactor programs. A sampling of the wealth of fuel-pin data that is currently available is given in table VII. Most fuel pins are bulk UO_2 with a cladding of stainless steel or zirconium. In some cases the cladding is niobium, tungsten, or molybdenum. In general, the surface temperatures are rather low

TABLE VII. - PIN FUEL DATA

Material	Surface temperature		Center temperature		Burnup, percent of UO_2	Source
	$^{\circ}F$	$^{\circ}K$	$^{\circ}F$	$^{\circ}K$		
Bulk UO_2	----	----	3000	1920	1.1	Atomic Energy Research
			3500	2200	.4	Establishment, England
Bulk UO_2 (Nb)	2400	1600	3600	2250	3.3	Battelle Memorial Institute
Bulk UO_2 (Nb)	1400	1030	3400	2150	3.1	Battelle Memorial Institute (MCR)
Bulk UO_2 (W)	4900	3000	>5000	>3000	.2	Battelle Northwest
Bulk UO_2 (Mo)	<500	<530	>5000	>3000	4.0	
Bulk UO_2	550	560	2700	1750	.4	Chalk River Reactor, Canada
Bulk UO_2	900	750	3500	2200	.2	
Bulk UO_2 - PuO_2 (stainless steel)	1000	810	3600	2250	13	General Electric (Valecitos)
Bulk UO_2 (Zr)	600	590	>5000	>3000	2.0	
Bulk UO_2	900	750	2000	1370	1.1	Oak Ridge National Laboratory
Bulk UO_2	1250	950	3000	1920	.3	
Bulk UC	2100	1420	2700	1750	2.5	Pratt and Whitney Aircraft Corp.
Bulk UN	2100	1420	2700	1750	2.8	
Bulk UO_2	600	590	2100 - 2300	1420 - 1530	12	Westinghouse Atomic Products
Bulk UO_2	600	590	2300 - 3200	1530 - 2000	6	

because most of the pins are designed for reactors which heat water to produce steam. In several cases, the center temperature is beyond the melting point of UO_2. In the case of bulk UO_2 pins, the burnup is limited by the amount of void designed into the pin to prevent excessive pressure buildup. A niobium fuel pin has been operated at a surface temperature of 2400° F (1590° K) with a burnup of 3.3 percent. Some bulk uranium carbide and uranium nitride pins have been run at temperatures of 2100° F (1420° K) with burnups of 2.5 and 2.8 percent, respectively. Two sets of data indicate burnups beyond 10 percent. However, these data were for surface temperatures of 1000° F (810° K) or less. Fuel pins have not been tested to any great extent with surface temperatures above 2000° F (1370° K).

As in the case of cermet fuel, all sources of fuel pins were thoroughly explored. Many companies can produce pins of our specifications, but there is a lack of test data in the temperature range and burnups of interest. Bulk UO_2 molybdenum fuel pins are presently being procured from two vendors prominent in this field. In the meantime, three different fuel pins have been fabricated in-house and irradiated in capsules similar to the capsule previously described. The results of these irradiations are shown in table VIII. Two of the fuel pins used a cermet fuel in place of bulk UO_2 as the fuel material within the pin. The burnups achieved at 2700° F (1750° K) were 1.5, 10, and

[redacted]

TABLE VIII. - LEWIS MOLYBDENUM

PIN IRRADIATIONS

Fuel material	Temperature		Burnup, percent of U-235
	^{o}F	^{o}K	
Cermet fuel	2700	1750	~1.5
Bulk UO_2	2700	1750	10
Cermet fuel	2700	1750	~6

6 percent for the three pins irradiated thus far. It should be possible with fuel-pin-type elements to achieve burnups well in excess of 10 percent. Additional fuel specimens are being irradiated as they are delivered. We anticipate that, during this next fiscal year, burnups greater than 10 percent can be demonstrated using state-of-the-art fuel-pin construction techniques.

One of the problems in fuel-pin design is the lack of knowledge of creep and stress rupture strength of refractory-metal tube materials. This knowledge is required to achieve long life. Generally, creep and stress-rupture data, when available, are obtained in sheet or bar stock form. These data are sensitive to the fabrication technique and, in many cases, to the direction in which the force is applied. Accordingly, the strength of the tubes was measured by actually pressurizing sample tube materials to provide stresses as they would occur in a fuel pin at the end of pin life when the internal pressure would be a maximum. The high-temperature tube test apparatus is shown in figure 14. Four tubes pressurized from high-pressure gas bottles are heated in a vacuum furnace to the temperature of interest. The tubes are pressurized in a fixed-volume system to pressures to 2000 psi (13.8×10^{6} N/m^{2}). The time it takes for the tube to creep and develop a leak, as indicated by a fall-off in pressure, is defined as the life. Attempts were then made to correlate the data with conventional sheet and rod material data. The data obtained from this test program for molybdenum tubes and for titanium (0.5 percent)-zirconium (0.08 percent)-molybdenum (99.25 percent) tubes are compared with bar and sheet handbook data in figure 15. The 500-hour creep-rupture stress is plotted as a function of temperature. Bands are shown to indicate the wide range of reported data. The 5000-hour Lewis creep-rupture stress data were extrapolated from 100- to 500-hour tests by means of the Manson and Haferd or Larson-Miller parameters (ref. 7). The handbook data were extrapolated in similar fashion, but the test data, in general, are for times much less than 100 hours. The Lewis data are about a factor of 2 lower than the handbook data. In terms of operating life at the same temperature, this difference amounts to a reduction of about an order of magnitude. Experiments are continuing to establish allowable long-time stresses for

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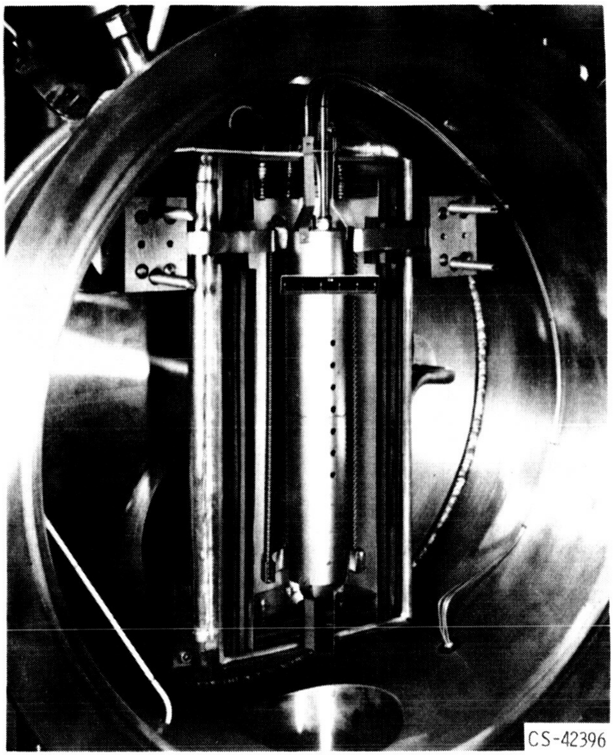

Figure 14. - High-temperature tube test apparatus.

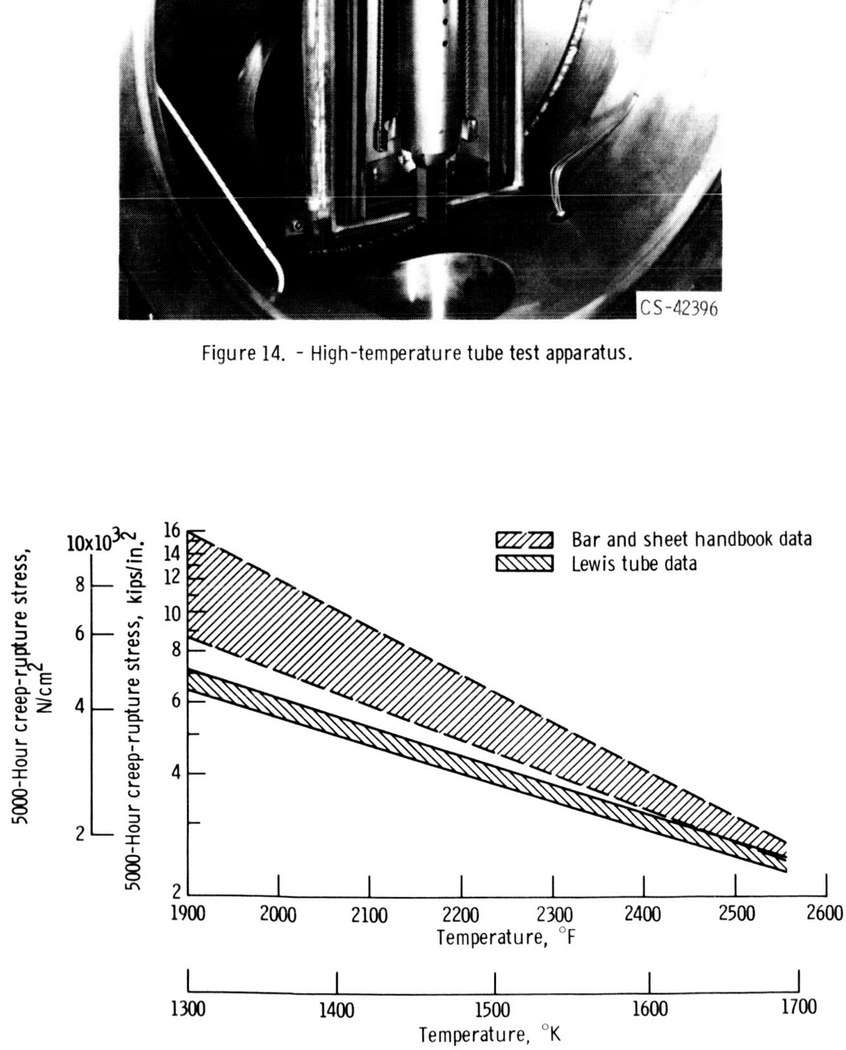

Figure 15. - Refractory tube tests of molybdenum and titanium-zirconium-molybdenum tubes.

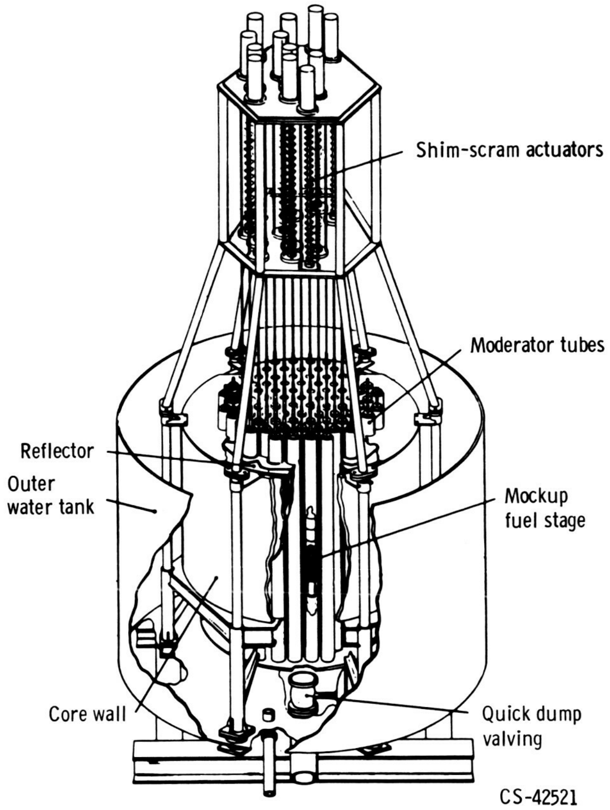

Figure 16. - Critical experiment performed by General Electric Company for AEC.

molybdenum fuel pins in both tension such as occurs at the end of fuel-pin life and compression such as occurs at the start of fuel-pin life for high-pressure reactors.

Long-life reactor structure. - Core support materials will be subject to large amounts of reactor radiation, which could affect properties such as strength and ductility. The core materials will also be subject to mass transfer and corrosion in those systems that use liquid metals. An area which needs further study is the use of high-density structural materials in the core. High-density core structure materials could serve as shield materials as well as performing their primary function. It would be quite advantageous from the point of view of minimizing reactor-plus-shield weight to fabricate reactor-core components of heavy metal alloys containing, for example, large amounts of tungsten.

Reactor neutronics. - Reactor neutronics studies are being conducted

(1) To verify reactor calculation techniques

(2) To determine reactor composition and dimensions

(3) To minimize peak-to-average fuel burnup

(4) To minimize reactor - shield-assembly weight

Analytical reactor calculation techniques must be verified to give confidence to the large number of reactor-core calculations which must be made in this study. Reactor-core calculations determine reactor compositions and dimensions and give power distributions within the core, which are necessary to investigate methods for minimizing peak-to-average fuel burnup. If all the fuel in the core reached the same maximum burnup, maximum core life would be achieved. In other words, the life of the core is as long as the life of the fuel element that had the maximum burnup.

There is an optimum distribution of reactor and shield materials which would minimize reactor - shield-assembly weight. For example, by placing heavy materials within the core, core weight increases. The reactor plus shield weight, however, may decrease because of the self-shielding increase within the core. Many calculations are required for trade-offs such as this to minimize the weight of the reactor-shield combination.

A series of critical experiments was performed at the AEC National Reactor Testing Station, Idaho to verify our calculation technique. Figure 16 shows the critical experiment performed by the General Electric Company which operate the test facility for the AEC. The experiment was for the water-moderated heterogeneous molybdenum-UO_2 reactor core. This core is probably the most difficult to calculate. The experiment (fig. 17) was a mockup of a conceptual design of a water-moderated molybdenum-UO_2 reactor in which the water fraction and the fuel fraction were parametrically varied.

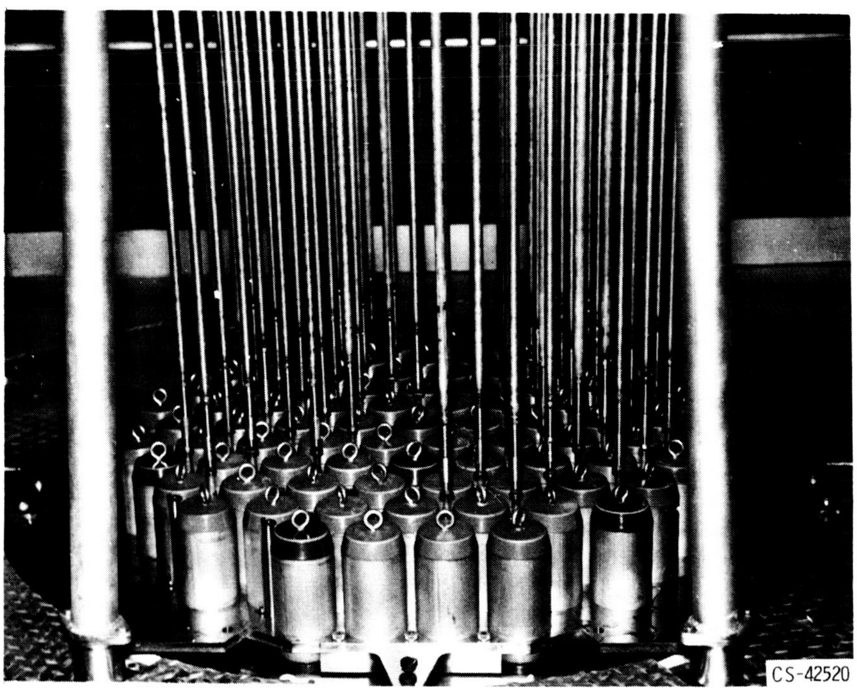

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Figure 17. - Critical experiment closeup.

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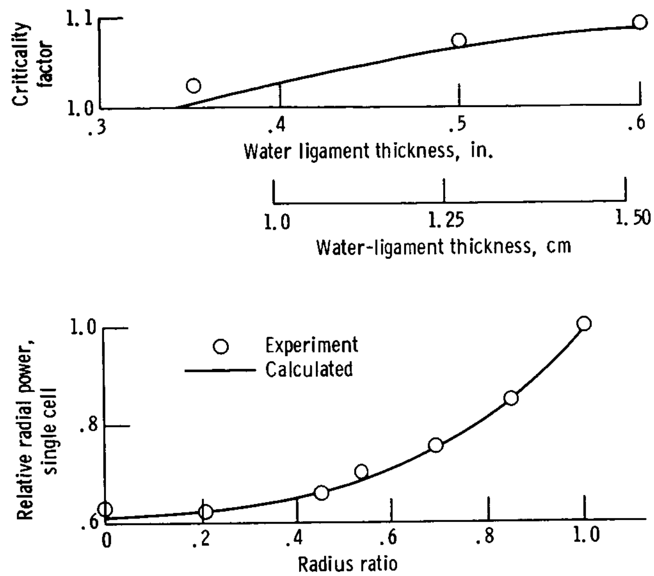

Figure 18. - Results of critical experiment with molybdenum-uranium-water core.

The results of the experiment, both with regard to predicting the multiplication factor and the power distributions in the core and within individual elements, agreed well with analytical predictions, as indicated in figure 18. In addition an experiment was conducted in which a major zone of the core was poisoned to simulate a control device that attempts to minimize peak to average burnup. The analytical techniques developed at Lewis proved adequate for our conceptual studies.

Long-Life Heat Exchangers

As in the case of the reactor, the operating life of the heat exchangers should be comparable to engine overhaul life. In other words, operating life should approach 10 000 hours. In the case of liquid-metal heat exchangers, there has been a great deal of experience in the past because of the interest in liquid metals for the ANP program and for the space power programs which followed. In addition there has been a great deal of experience in connection with fast-breeder power reactor programs. There is less experience in the use of high-pressure inert-gas heat exchangers, which are of interest for the helium systems.

Liquid-metal heat exchangers. - Examples of liquid-metal-to-liquid-metal heat exchangers that were operated for the Lithium Cooled Reactor Experiment (LCRE) (ref. 8) are shown in figure 19. All the components in a multiloop system of which this exchanger was a part were operated for 10 000 hours. The parts were fairly large scale, as reflected by the 5-megawatt operating power level. For a lithium-to-lithium heat exchanger made of columbium - 1-percent zirconium (fig. 19(a)), the hot-side lithium

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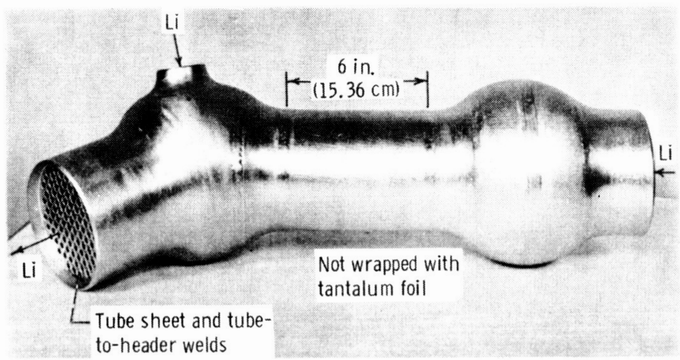
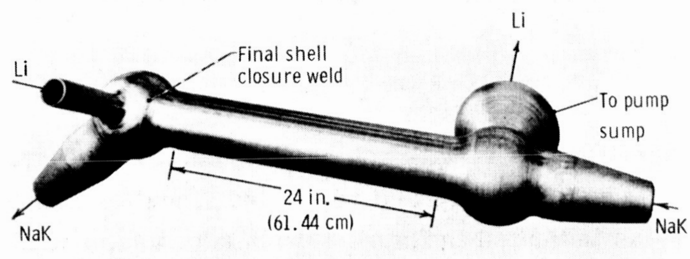


(a) Lithium-to-lithium heat exchanger. Material, niobium - 1-percent zirconium; primary inlet temperature, 1950° F (1340° K), primary outlet temperature, 1620° F (1160° K); secondary inlet temperature, 950° F (780° K); secondary outlet temperature, 1350° F (1000° K).



(b) Lithium-to-NaK heat exchanger. Material, niobium - 1-percent zirconium; primary inlet temperature, 1350° F (1000° K); primary outlet temperature, 950° F (780° K); secondary inlet temperature, 700° F (650° K); secondary outlet temperature, 1250° F (950° K).

Figure 19. - Liquid-metal heat exchanger tests for liquid-cooled reactor experiment (LCRE). Operating time, 10 000 hours; operating power, 5 megawatts.

inlet temperature was 1950° F (1340° K). The cold-side outlet lithium temperature was 1350° F (1000° K). The heat exchanger operated satisfactorily for the entire duration of the test.

The 1350° F (1000° K) lithium from the primary heat exchanger served as the hot fluid for a lithium-to-NaK heat exchanger (fig. 19(b)). The NaK outlet temperature was 1250° F (950° K). This heat exchanger is similar to the liquid-metal-to-liquid-metal heat exchanger for a liquid-metal aircraft system. The heat from the NaK circuit was rejected to air by means of a NaK-to-air heat exchanger shown in figure 20. This heat exchanger also ran satisfactorily for the full test duration of 10 000 hours. The temperature of the liquid metal entering this heat exchanger was 1250° F (950° K).

Extensive testing was carried out at the Oak Ridge National Laboratory on NaK-to-air heat exchangers during the ANP program. A typical NaK-to-air heat exchanger that was tested at Oak Ridge is shown in figure 21. A great deal of data has been obtained in a temperature range from 1400° to 1600° F (1033° to 1144° K). At least one of the heat exchangers was operated for 1200 hours at a time when this was about a factor of 10 longer than was considered necessary. Emphasis in the tests was placed on thermal

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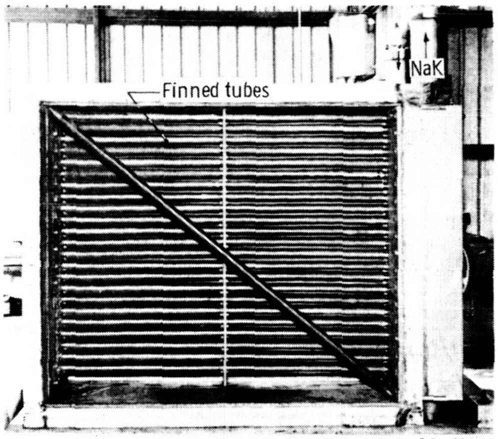

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Figure 20. - NaK-to-air heat exchanger for liquid-cooled-reactor experiment. Operating time, 10 000 hours; operating power, 5 megawatts; material, 316- and 320-stainless steel; primary inlet temperature, 1250° F (950° K); primary outlet temperature, 700° F (650° K); secondary inlet temperature, 85° F (300° K); secondary outlet temperature, 400° F (480° K).

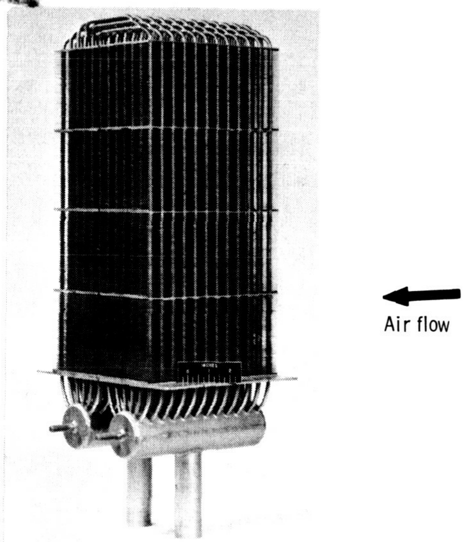

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Figure 21. - Typical NaK-to-air heat exchanger test at Oak Ridge for ANP program. Material, 310- and 316-stainless steel; NaK inlet temperature, 1400° to 1600° F (1030° to 1140° K); operating time, greater than 1200 hours.

cycling. The test heat exchanger that ran for 1200 hours was cycled from hot to cold more than 50 times during its test. No data were obtained in the ANP program operating times approaching 10 000 hours. Thermal cycles are probably a serious detriment to long-life performance and warrant careful consideration in design followed by experimental checkout and final development.

Fabrication of heat exchangers involved in the liquid-metal system seems to be well developed. Operating experience at times of interest to aircraft nuclear systems has been demonstrated. Data are lacking, however, at temperatures of interest for refractory-metal systems. Benefiting from the use of refractory metals requires NaK temperatures into the air heat exchangers of at least 1500° F (1090° K), which is about 250° F (390° K) higher than the test previously described. One area which deserves further attention in the liquid-metal systems is the embrittlement of refractory fuel elements in the presence of nitrogen and/or oxygen that has been transported through the heat exchangers to the fuel elements themselves. The use of niobium in aircraft liquid-metal systems is not considered possible at present for operating times greater than 1000 and 2000 hours and for a liquid-metal temperature into the air heat exchangers of 1500° F (1090° K). Tantalum may have better promise than niobium in this regard but few data are presently available. Data are now being obtained for tantalum in connection with space power programs. An additional refractory-metal problem area is the protection of the external portions of the refractory-metal system, such as the reactor vessel, heat-exchanger shells, ducting, and valves, from air contamination by the use of vacuum jacketing or ultra-high-purity inert-gas jacketing. The 10 000-hour system previously

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described was operated in a vacuum with tantalum foil wrapping used as a getter for any oxygen or nitrogen that diffused or leaked through the jacket walls.

Based on our technology surveys to date, it appears that the highest metal temperature in all stainless-steel liquid-metal systems is limited to about 1250° to 1300° F (950° to 980° K) for 10 000-hour operation with corrosion penetration less than 5 mils (0.13 mm). If the maximum fuel-element temperature is set at 1300° F (980° K), then the liquid-metal temperature leaving the reactor would be about 1150° F (890° K). Allowing only a 50° F (28° K) drop in temperature through the liquid-metal-to-liquid-metal heat exchanger results in an 1100° F (870° K) liquid-metal temperature entering the air heat exchanger. The corresponding turbine inlet temperature would be only about 950° F (780° K). Molybdenum fuel elements could most probably be substituted for the stainless-steel elements, inasmuch as molybdenum does not suffer from the severe nitrogen and oxygen embrittlement problem as niobium and tantalum do. In this case, the liquid-metal temperature leaving the reactor could be 1300° F (980° K) which would give a turbine inlet temperature of about 1100° F (870° K). (Our optimization studies have shown that reasonably good performance might be expected with turbine inlet temperatures in this range.)

At present, the liquid-metal system which appears to us to have the most promise of achieving 10 000-hour life with little corrosion and with a turbine inlet temperature within reason is a system which uses molybdenum fuel elements in an otherwise all stainless-steel system in spite of the fact that this specific system has not been experimentally verified. The turbine inlet temperature would be expected to be no higher than 1100° F (870° K). For liquid metal systems, the all stainless-steel - molybdenum fuel-element system is felt to have the most promise for application to long-life aircraft propulsion systems for two reasons. First, it probably would produce a turbine inlet temperature just as high as the best demonstrated refractory metal system which has heated air. Second, a stainless-steel system is felt to be a more practical, reliable, and lighter weight system than a refractory system when the problems of air environment are faced in practice. Refractory metals are highly attractive for space systems because they can be operated at high temperatures in the vacuum environment that exists. This is contrasted with the use of refractory metals for atmospheric systems which are designed to heat air where extraordinary measures are required to protect the system from any trace of air. This study of liquid-metal technology is being continued to firmly establish proper conclusions on the potential of liquid-metal systems.

High-pressure inert-gas heat exchangers. - Little work has been done on high-performance high-pressure gas-to-air heat exchangers. The helium propulsion systems discussed in this report require the heat exchanger which heats the air of the turbofan engine to operate at pressures in the range of 1000 to 2000 psi (6.9×10^{6} to 13.8×10^{6} N/m^2). The operating temperatures should be as high as possible without exceeding material strength and oxidation limits and without exceeding overall heat-exchanger size and weight limitations. The heat-transfer surface temperature should not be less than about 1250° F (950° K) if this system is to be competitive with the liquid-metal system. This section includes a discussion of various aspects of high-pressure gas-to-air heat-exchanger problems and some of our studies to demonstrate potential performance and feasibility. In particular, strength and fabrication studies of oxidation-resistant heat-exchanger tube and header materials are discussed.

The primary materials problem in an inert-gas-to-air heat exchanger is the requirement for high-temperature strength combined with oxidation resistance. Because of the lack of data on high-pressure gas-to-gas heat exchangers, some basic experimental and analytical work was performed to determine the problems and performance potential for these exchangers. A survey was made of high-temperature oxidation-resistance materials that could be fabricated into tubes. Tubes made of all known high-temperature oxidation-resistance materials were purchased and tested in the tube test

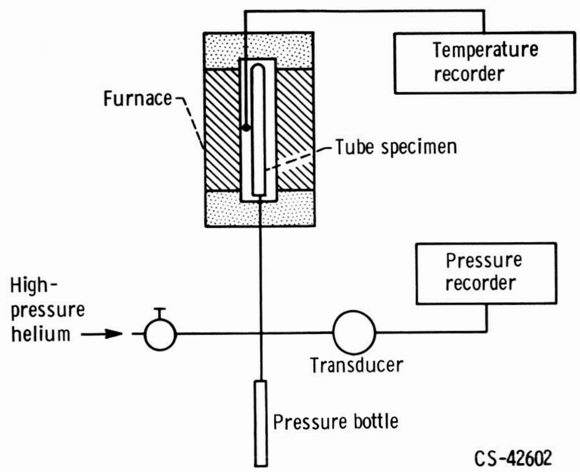
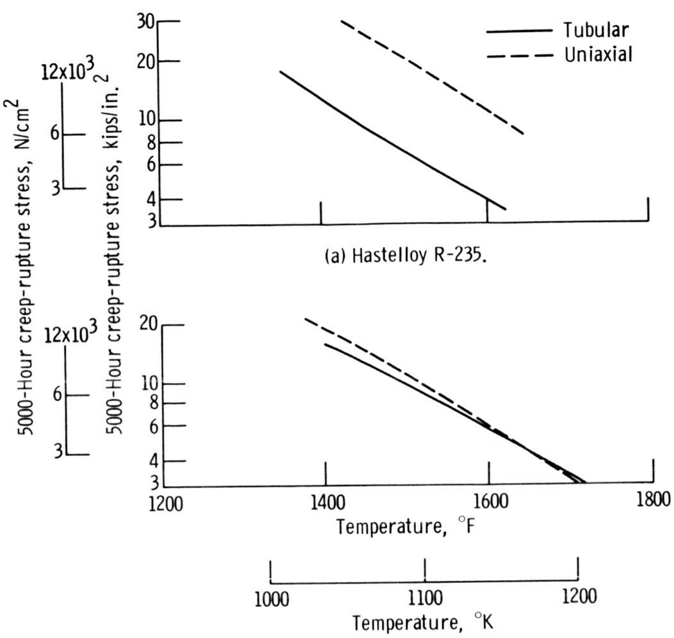


Figure 22. - Tube test apparatus.

rig shown schematically in figure 22. The tubular specimen is mounted in a constant-temperature furnace and pressurized by a source of high-pressure helium. The pressure is recorded as a function of time. The time it takes to develop a leak in the specimen is defined as the tube life. As in the case of fuel-pin tubes, creep-rupture stress has been correlated with earlier handbook values. The sample results of our tests are shown in figure 23. The 500-hour creep-rupture stress is plotted as a function of temperature for 2 of the 15 or so high-strength oxidation-resistant materials being tested. The handbook uniaxial data are shown as the dashed line for both Hastelloy R-235 and N-155 (Multimet). The data obtained from our tube tests are shown as the solid line.



(b) N-155 (Multimet).

Figure 23. - Tube test data.

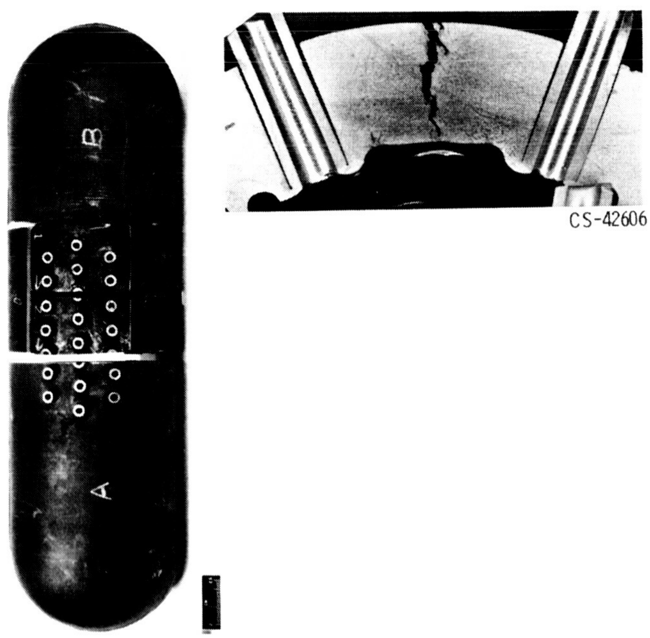


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Figure 24. - High-pressure gas-to-air heat-exchanger test. Temperature, 1550° F (1120° K); pressure, 1500 psi (1090 N/m^2); design life, 1000 hours; actual life, 2567 hours.

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In the case of N-155 the agreement between the two sets of data is fairly good. The data for the Hastelloy R-235 are indicative of the difficulty in correlating uniaxial with triaxial data such as are obtained from stressed tubes. It is clear that more data are required for reliable design work. Tentatively, N-155 is considered the best material for high-pressure heat exchangers considering the factors of good strength, ductility, weldability, fabricability, and availability. Better materials may be available in the future.

The tube test apparatus shown in figure 22 will also be used to test tube to header joints. Single tube to header joints have been made and will be operated under pressure until failure (leakage) occurs. This program will be used to study existing techniques for making joints with the best tube materials. The objective is to find joints which will permit the tubes to operate at full capacity. These tests are now in progress.

In addition to individual tube tests a heat-exchanger header test was conducted at Oak Ridge. This work is illustrated in figure 24. A full-scale section of helium-to-air heat-exchanger header was fabricated with enough tubes to duplicate the stresses that would occur in the real heat-exchanger header. The test section was pressurized at 1500 psi (1×10^{7} N/m^{2}) with a constant-volume system and operated at a temperature of 1550° F (1120° K) for over 2500 hours before failure occurred. Failure is defined as a decrease in gas pressure. The test was designed for 1000 hours of operation. The material used was N-155, which appeared to be the most promising from the tube tests. A photograph taken after the test (fig. 24) shows that the header failed in the ligament region between the tubes, as was predicted. Also shown is that the tube-to-header joints were sound after the test. Neither the header tests nor the tube tests gave any indication of explosive failures, which might be anticipated because of the high pressure involved. The failures all occurred as leaks that increased gradually over a period of several hours.

Leaktightness may be a problem, not because of the feasibility of making a single leaktight joint, but because there are many joints that must remain leaktight during thermal cycling and aircraft and engine operating environments. The chief problem becomes one of quality control, inspection, and reliability demonstration. A facility has been designed for demonstrating the leaktightness, structural integrity, and heat-transfer characteristics of representative sections of helium-to-air heat exchangers. Construction and operation of this facility awaits future funding. At this time we feel that we are able to design a helium-to-air heat exchanger that is reasonable in weight and provides long operating life. What remains to be done is to demonstrate this conclusion experimentally at simulated aircraft operating conditions.

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PROPULSION SYSTEM STUDIES

Conceptual designs are being studied in order to properly assess the potential performance of nuclear powerplants. The studies are sufficiently detailed to provide realistic weight estimates. As information is gathered from technology surveys and from the results of the analytical and experimental studies, the conceptual reactor and powerplant designs are continually updated. For example, it is not possible on the first round of design studies to integrate safety considerations and shielding, reactor, and shock-absorbing structures. As information is gained in each of these areas, they are incorporated into the designs. The final designs will therefore not be complete until the end of the program.

A technology survey and a first-round conceptual reactor - shield-assembly design has been carried out for each of the three following candidate propulsion systems. The studies were done by contract as indicated by the contractor's name in parenthesis after each system:

(1) Fast-reactor liquid-metal-cooled system (General Motors Allison)

(2) Fast-reactor gas-cooled system (General Electric-NMPO)

(3) Thermal-reactor gas-cooled system (Aerojet-General Nucleonics)

These studies have been completed. The final report for the thermal gas-cooled system by Aerojet-General has been accepted. Drafts of the final reports of the fast liquid-metal and fast gas-cooled systems are being reviewed and edited before final acceptance.

It is planned to carry out second round studies called integrated powerplant designs for all three systems. These studies will incorporate our latest information on shielding and safety provisions, including containment vessels, quick-acting containment valves, shock-absorbing equipment, and provision for the high-acceleration loads that would occur on impact.

In addition to the first- and second-round propulsion-system studies, the problem of integrating the reactor-shield and containment-vessel package with an impact-energy-absorbing system structure and with the aircraft structure is being investigated. The purpose is to determine a realistic weight estimate of a shock-absorbing system that is integrated with the aircraft and reactor-shield assembly.

Finally, optimization studies are being performed to provide values for the operating values for the operating variables of each component of the powerplant and aircraft which maximize overall aircraft performance. These data are used to select operating conditions for the more detailed conceptual design studies. The input data and the component performance maps for the optimization studies are contiually updated from the knowledge gained from the technology programs.

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First-Round Conceptual Design Studies

The major results of each of the first-round conceptual design studies are presented in the following paragraphs.

Liquid-metal-cooled fast-reactor system. - In this study, Allison was asked to first make a complete liquid-metal technology survey. With this as a basis, they were to make conceptual designs of two systems. The first was to be an all stainless-steel system since the most experience has been obtained with this system. Secondly, they were to make a conceptual design of an advanced system using materials such as refractory metals which are not as well developed but have good potential, as supported by some experimental indications. Figure 25 shows the conceptual layout of the more advanced

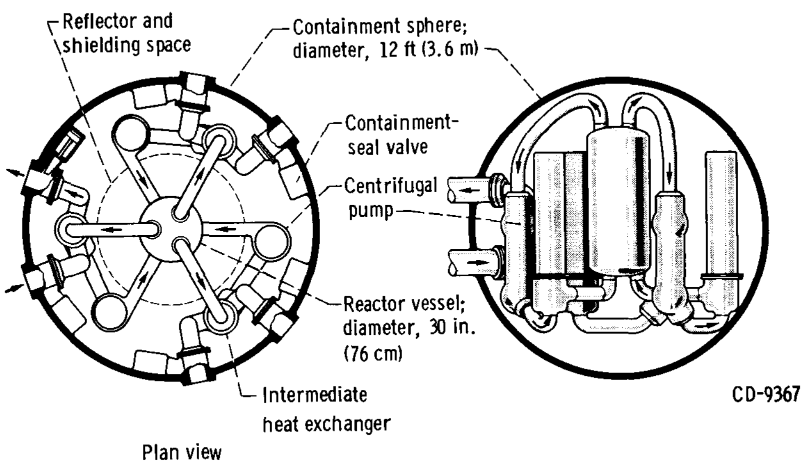

Figure 25. - Liquid-metal-cooled fast-reactor system.

refractory-metal system. The reactor is composed of fuel pins consisting of tantalum cladding filled with UO_2. The reactor primary coolant is lithium. The lithium is circulated from the reactor core to the liquid-metal-to-liquid-metal heat exchanger by means of a canned rotor pump. The lithium heats sodium that is circulated to the engine liquid-metal-to-air heat exchangers. Inasmuch as it was specified that no radioactive material be permitted outside the containment vessel, the primary lithium loop had to be completely contained within the containment vessel. This loop is divided into three parallel systems. Each has its own canned rotor pump and liquid-metal-to-liquid-metal (lithium-to-sodium) heat exchanger. Fast-acting emergency shutoff valves are incorporated in each of the three outlet and inlet lines for the secondary sodium system. The valves as proposed by Allison are gate valve with tapered seats that are driven by gas

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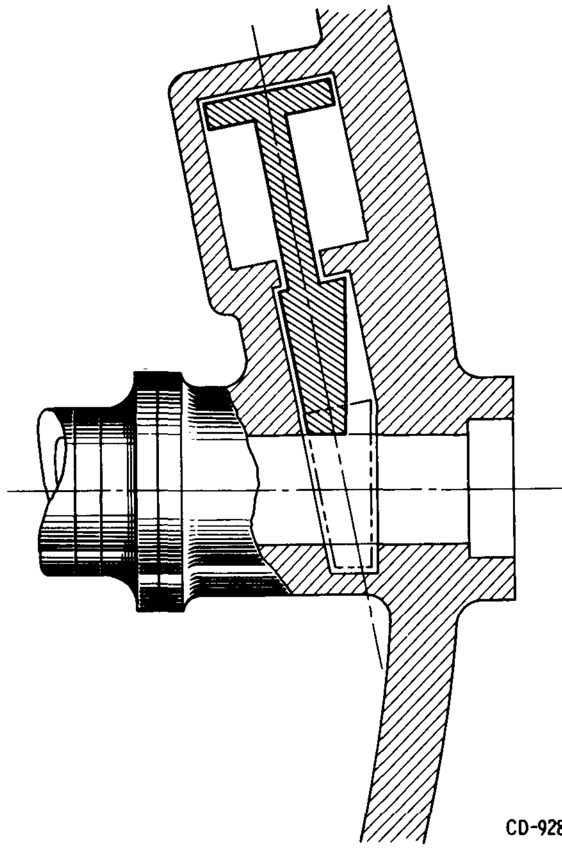

Figure 26. - Fast-acting gate-type containment valve.

pressure cylinders (see fig. 26). The valves are operated by a suitable crash-detection device (for example, strategically located accelerometers).

The reactor power is 250 megawatts and the diameter is about 30 inches (76 cm). The reactor outlet lithium temperature is 2000° F (1360° K). The sodium temperature at the inlet of the air heat exchanger in the powerplant is 1650° F (1170° K). The turbine inlet air temperature is 1500° F (1090° K). Reactor control is provided by rotating reflector drums. Gamma and neutron shielding is provided outside the reflector. The shielding is composed of alternate layers of tungsten and water. The smallest containment sphere that could house the reactor and primary-loop equipment was 12 feet (3.7 m) in diameter. The containment-sphere material is a tantalum alloy, such as T-111 or T-222.

After this study was initiated, it became apparent that a shock-absorbing system can probably be designed which could prevent the containment vessel from contacting the Earth, as discussed in the section Fission-product containment. Such a system would allow air to circulate freely around the containment vessel. Thus, there would be no need for a refractory containment vessel since the maximum temperature of the contain-

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ment vessel in air will be less than 1700° F (1200° K). This feature will be incorporated in the second-round study.

The preliminary weight estimate of this reactor-shield assembly is not yet complete, pending further shield calculations. There are two main difficulties with this high-temperature liquid-metal system. The first is that the bulkiness and awkwardness of the components that must be contained within the containment vessel require it to be 12 feet (3.7 m) in diameter. At present, we feel this system would be excessively heavy. Further work must be done to make a more compact system. Secondly, there is much uncertainty (as discussed in the section Liquid-metal heat exchangers) with regard to refractory-metal embrittlement due to diffusion of nitrogen and oxygen. The nitrogen and oxygen enter the system through the liquid-metal-to-air heat-exchanger walls and are transferred from the sodium to the lithium through the refractory-metal walls of the intermediate heat exchanger. At this point the fuel elements which are at the highest temperature in the system absorb them and become embrittled. The effect of this problem on reactor life, particularly considering the large number of thermal cycles which will be required during the life of the powerplant, has not been assessed conclusively. In an attempt to solve the embrittlement problem, Allison has proposed the use of a bimetallic clad sodium-to-air heat exchanger to minimize the diffusion of oxygen and nitrogen. Further work is necessary to demonstrate the feasibility of this approach.

The stainless-steel conceptual design also needs further work. The results when completed should be similar to the high-temperature system with regard to weight but should produce a turbine inlet temperature no higher than 1100° F (870° K), as discussed in the section Liquid-metal heat exchangers. Although this turbine inlet temperature sounds low, optimization studies indicate that acceptable performance of the aircraft can be obtained at this temperature. The results of the technology survey that was made by Allison and the first-round attempts at designing liquid-metal systems were examined at Lewis. In our opinon, there is, at present, no data which support the use of niobium nuclear powerplants with lifetimes approaching 10 000 hours unless the turbine inlet temperature is about the same as that which can be obtained by stainless-steel systems using molybdenum fuel elements. Tantalum, although it has similar problems, has greater promise than niobium. It is apparently somewhat less susceptable to nitrogen and oxygen diffusion and is a stronger material. It is not possible to determine the performance with tantalum because of the lack of corrosion and mass-transfer data and because there is no fuel-element data at present.

A further problem, which has not been discussed, is that liquid-metal systems require that major rotating machinery be included in the unaccessible reactor compartment. The potential problem here is to determine the probability of high reliability for unattended rotating liquid-metal machinery. The latter problem may not be serious con-

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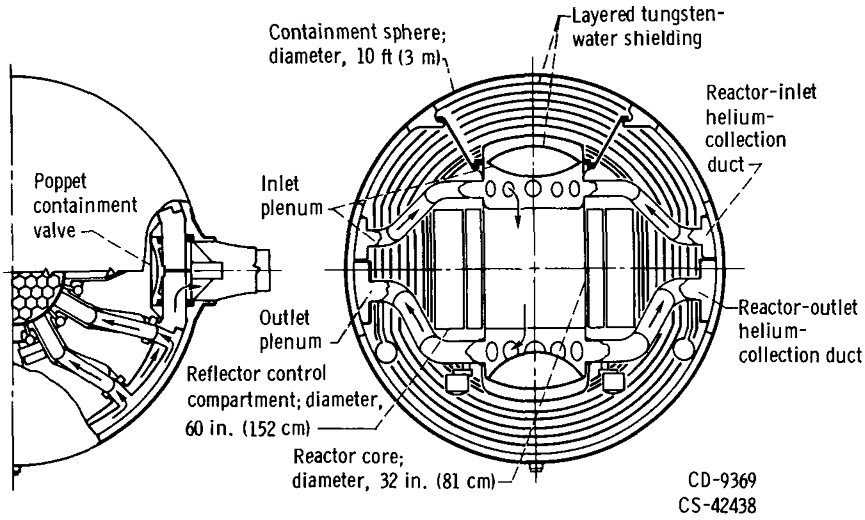

Figure 27. - Helium-cooled fast-reactor system.

sidering that ORNL has operated centrifugal liquid-metal and molten-salt pumps for an accumulated total of 450 000 hours in the temperature range from 1000° to 1300° F (810° to 977° K) (ref. 9).

Helium-cooled fast-reactor system. - The gas-cooled fast-reactor concept as conceived by General Electric (NMPO) is shown in figure 27. The reactor core is 32 inches (81 cm) in diameter and is composed of an array of GE-710-type reactor fuel elements. (ref 10). These fuel elements are composed of a tungsten-UO_2 matrix clad with a tungsten-rhenium-molybdenum alloy. These elements have been under investigation for the GE-710 reactor for a number of years. The fabrication techniques are fairly well worked out, and fuel burnup characteristics are being established. As indicated earlier, 1-percent fuel burnups have been demonstrated in this type of fuel element at temperatures somewhat above those probably required for aircraft reactors. The design reactor lifetime was 2000 hours. Indications are that a lifetime of 5000 hours is possible with some minor changes in design.

The gas-cooled fast reactor has a plenum at both ends. The helium is carried away from the high-temperature plenum at the lower end of the reactor by multiple ducting with bends. The bends provide for thermal expansion and reduced streaming of radiation from the core. The multiple ducts empty into a tapered toroidal collection duct which encircles the inside surface of the containment vessel. The inlet plenum, the multiple-duct configuration, and the distribution ducts are similar to the outlet system. Quick-acting emergency shutoff valves are provided at the inlet and the outlet of the collection plenums. The valves in this case are poppet-type valves with serrated seats that have interference fits and provide positive seal when closed (see fig. 28). The serrated sur-

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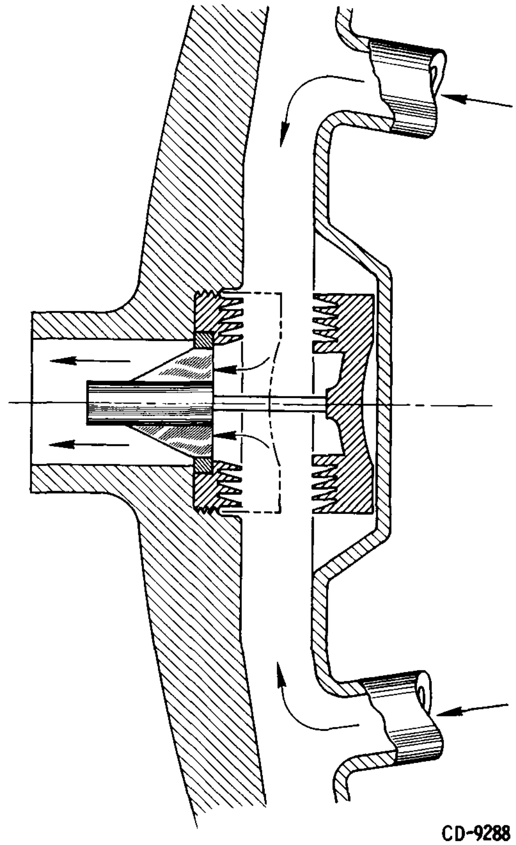

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Figure 28. - Fast-acting poppet-type containment valve.

faces could bc coated with materials that would weld due to the high surface pressures encountered during the rapid closure.

Reactor control is provided by rotating reflector segments away from the core. Shielding is provided by alternate layers of tungsten and water. The containment vessel is 10 feet (3 m) in diameter and made of T-111 or T-222 tantalum alloy. The comment with regard to the possibility of using a nonrefractory-metal containment vessel applies here, as well as in the previously discussed liquid-metal system. The second-round study will use a conventional pressure-vessel material for the containment vessel.

In this design the helium outlet temperature from the reactor is 1800° F (1250° K). The turbine inlet temperature using a conservative helium-to-air heat-exchanger design would be about 1400° F (1030° K). Turbine inlet temperatures higher than 1500° F (1090° K) may be possible. The maximum fuel-element surface temperature for this design is 2180° F (1470° K). The hot-spot temperature is about 2500° F (1640° K). The reactor power is 250 megawatts. An overall reactor - shield-assembly weight estimate for this concept has not been made. This system should probably weigh about the same as the thermal gas system which is discussed in the following section. The weight will

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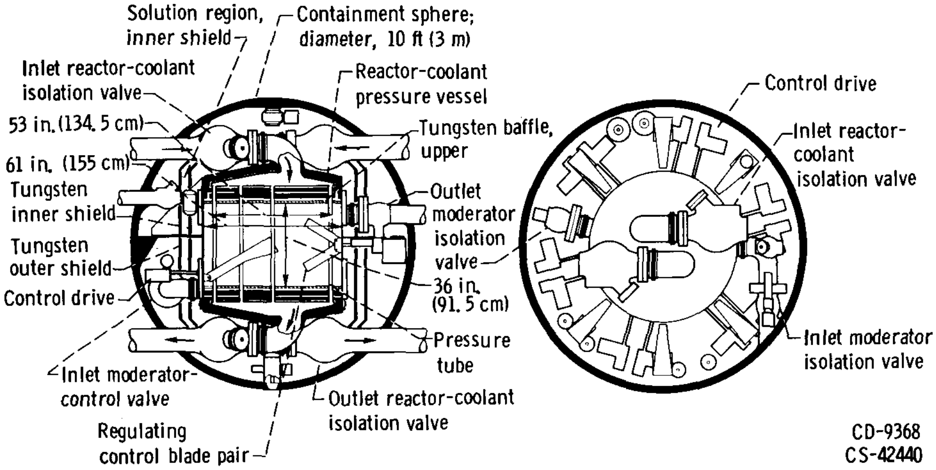

Figure 29. - Helium-cooled thermal-reactor system.

probably be in the range of 200 000 to 250 000 pounds (91 000 to 113 000 kg). Further shield calculations are required to make a firmer estimate.

Helium-cooled thermal-reactor system. - The results of the first-round helium-cooled thermal-reactor concept study as completed by Aerojet-General Nucleonics is illustrated in figure 29. The reactor core is an outgrowth of the ML-1 reactor concept that was designed, built, and operated by Aerojet for the Army. The reactor core is water moderated, and is essentially a tank of water which is pierced with aluminum or stainless-steel tubes so that the water is contained outside these tubes. Into the tubes are inserted fuel elements which are composed of clusters of molybdenum-clad UO_2 pins. The coolant gas is heated as it flows along the outside surfaces of these pins to its final reactor outlet temperature. The helium is collected in a plenum at the reactor outlet. Two reactor outlet coolant lines penetrate the 10-foot (3-m) containment sphere. In this case the quick-acting shutoff valves are flapper-type valves which do not seem promising compared with the wedge-type and poppet-type valves discussed in the previous concepts. The Aerojet study was made earlier than the others and the valves and containment-vessel requirements were added to the study after the start of the contract for the sake of completeness rather than as a result of a careful study. In the short time available for the completion of the contract, valve vendors were not able to supply Aerojet with a reasonable design other than the conventional valve shown.

In this concept, as in the others, the entire volume within the containment sphere except for the gas passages is water filled to provide neutron shielding between the tungsten gamma shields. Reactor control is provided by means of semaphore-type control blades that penetrate the reactor core between the pressure tubes. The helium outlet temperature, the fuel-element operating temperatures, the hot-spot temperatures, and the turbine inlet temperatures for the thermal-reactor concept are the same as for the fast reactor. The overall reactor assembly diameter including reflectors is the same as for the General Electric fast helium system. The reactor packages are therefore interchangeable and either ducting system can be used with either reactor.

A preliminary crude weight estimate, neglecting the shielding effect of the containment vessel, was made prior to detailed shield calculations and resulted in a weight of 340 000 pounds (154 000 kg) for the reactor-shield assembly. Inasmuch as this study was completed earlier than the other two, there was time for a detailed three-dimensional UNC-SAM2 shield calculation, which resulted in an overall weight of 284 000 pounds (129 000 kg) when the shielding effect of the containment vessel was included. The weight breakdown is shown in table IX. The structure that is referred to includes all piping, auxiliary equipment, valves, and other items contained between the reactor and the containment vessel that are not a normal part of the ordinary shield. The outer shield is approximately 2 feet (61 cm) thick and is entirely outside of the containment vessel. The containment vessel in this case provides a substantial amount of gamma shielding and accounts for the major difference between the preliminary crude and the final weight estimates, as previously mentioned.

TABLE IX. - REACTOR SHIELD

WEIGHT FOR GAS-COOLED

THERMAL REACTOR

Component	Weight	
	lb	kg
Reactor (dry)	22 300	10 100
Moderator water	1 200	550
Inner shield water and structure	27 400	12 400
Outer shield water	88 900	40 400
Containment vessel	87 600	39 800
Tungsten shield	56 700	25 750
Total	284 100	129 000

Integrated Impact-Energy-Absorption Systems

Integration of the impact-energy-absorption system, the containment vessel, and the airframe should result in weight savings because of the dual or triple functions that could be served by these materials. Only very preliminary studies have been made thus far in this area. Studies were made in which the mounting of nuclear powerplants in the

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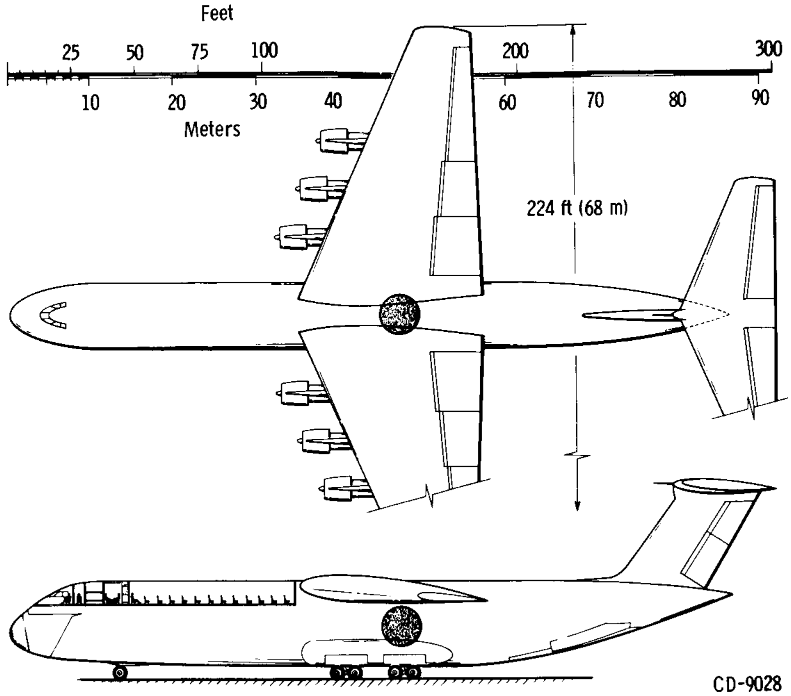

Figure 30. - Atmospheric nuclear transport system. Altitude, 36 089 feet (11 000 m); Mach 0.8; gross weight, 1 million pounds (454 000 kg).

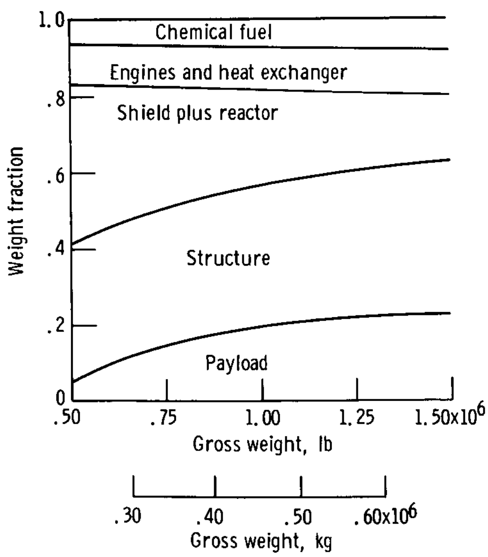

Figure 31. - Typical weight breakdown.

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C5A was considered. They indicate that the fuselage structure can be used to absorb substantial amounts of energy. In addition, a frangible-tube energy-absorption system might be incorporated as part of the basic aircraft structure to help reduce the weight penalty of energy absorbers.

Aircraft and Propulsion System Optimization Studies

Because of the large number of variable powerplant and aircraft operating parameters, some technique for choosing the best values was deemed necessary. In other words, some sort of optimization program was needed. Such a program would integrate all these variables and maximize airplane performance in terms of some figure of merit, such as payload or payload delivery rate. With the advent of high-speed computers, which were not available during the ANP program, a computer program was developed which simultaneously optimizes more than 15 aircraft and powerplant variables. This program contains the parametric characteristics of every major component of the powerplant and the aircraft. Each of the characteristics are given in terms of weight or thermodynamic or aerodynamic parameters, whichever criteria apply. For example, in the case of the aircraft the aspect ratio was optimized since there is a tradeoff between low wing weight with low aspect ratio and high aerodynamic efficiency with high aspect ratio. In the case of the engine, for example, weight and performance maps of the reactor coolant-to-air heat exchanger include consideration of the creep strength of the tube material as a function of temperature, pressure level, inlet and outlet temperatures, and pressure drop. The reactor-shield weight is a function of reactor size, power density, and power level. Allowances are made for the shield-weight increase due to duct penetrations. Each major component has been characterized by weight and performance maps which are updated as new information is gathered.

A sample optimized aircraft configuration, the major dimensions of which were determined by this program is shown in figure 30. This nuclear-transport-type aircraft with large cargo volume designed for a cruise speed of Mach 0.8 at an altitude of 36 000 feet (11 km) has a gross weight of 1 million pounds (454 000 kg). The reactor-shield assembly is located near the center of gravity, as would be expected. The six engines, mounted on the wings, are ducted-fan engines with an optimized bypass ratio of about 2. The wing optimized at the aspect ratio shown. The overall airplane lift to drag ratio for maximum payload is about 16 to 17. Cargo space is provided fore and aft of the reactor assembly. The engines are designed so that chemical augmentation for takeoff, climb, and landing is supplied by means of burners between the heat exchanger and turbines. The takeoff and climb performance is similar to the C5A. The wing span is 224 feet (68 m) and the overall length is just under 300 feet (92 m).

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A typical weight breakdown is shown in figure 31. The cruise speed is Mach 0. 8 at an altitude of 36 000 feet (11 km). The weight fraction is plotted as a function of the gross weight in millions of pounds ($kg \times 10^6$). The top block is the allowance for chemical fuel which amounts to 6 to 7 percent and is adequate for one landing, one takeoff, one climb, one descent, and a 1-hour cruise on chemical power. The engines include all rotating machinery, ducting, pumps, and heat exchangers and represents about 10 percent of the gross weight. The shield-plus-reactor weight includes and allowance of about 50 000 pounds (23 000 kg) for the weight of impact-energy-absorbing equipment. The shield-plus-reactor-weight fraction reduces substantially as the gross weight is increased. This decrease is a result of the fact that the shield-plus-reactor weight varies about as the square root of the gross weight or reactor power while most other components of the aircraft tend to vary more directly with the gross weight. The aircraft structure, for example, is about 35 to 40 percent of the gross weight and is essentially independent of the gross weight.

The decreasing shield-and-reactor-weight fraction with increasing gross weight results in an increase in payload with gross weight. We feel that the weights shown in figure 31 with the exception of energy absorber weight are realistic representations of current practice. For a gross weight of 1 million pounds (454 000 kg) the payload is about 20 percent of the gross weight. A million-and-a-half pound (680 000-kg) aircraft would carry 23 percent of its gross weight or 270 000 pounds (122 000 kg) of payload.

During a recent study (ref. 11) estimates were made of the technology that might be available 10 years from now. In figure 32, is shown the weight breakdown of a future nuclear aircraft assuming that these technology advances will occur. This weight breakdown is for a nuclear aircraft designed for Mach 0. 8 at 36 000 feet (11 km) with the assumption of the same fraction of chemical fuel as previously. It was estimated that engine weight would be reduced by 20 percent with normal advancements. In the case of the shield-plus-reactor weight, further refinement, shield shaping, improved shield materials and practical experience might reduce the weight by 30 percent. Thus, the shield-plus-reactor weight is estimated at 0. 7 of the weight shown in figure 31. In the case of the aircraft structure, it was estimated the structure weight would be 0. 6 of current practice. This leads to structure- to gross-weight ratios of about 0. 20, which could occur through the use of advanced materials such as composites. Even greater weight reductions than those shown here have been suggested. However, these values were chosen as the possible decrease in weight. The effect on payload is, of course, quite marked. Payload would be of the order of 40 percent of the gross weight. In other words, if all these advances could be realized, a million-pound (454 000-kg) aircraft would carry 400 000 pounds (180 000 kg) of payload, and a million-and-a-half-pound (680 000-kg) aircraft would carry 600 000 pounds (270 000 kg).

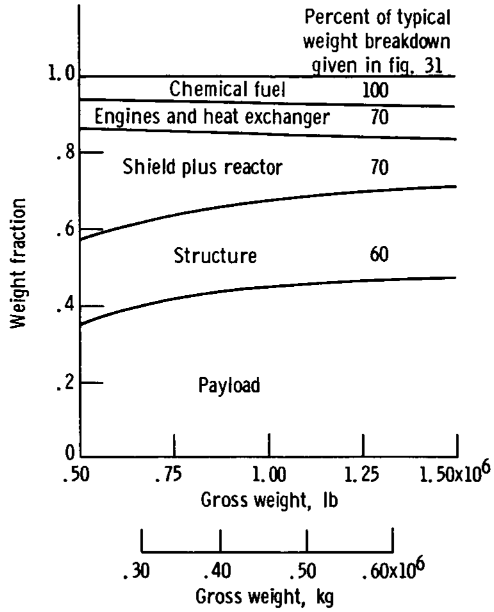

Figure 32. - Future weight breakdown.

SUMMARY OF RESULTS

The Lewis advanced technology study to assess the feasibility of safe, practical, publicly acceptable nuclear-powered aircraft was begun in August 1964. The study considers safety as the primary concern. Long reactor life, approaching 10 000 hours, is considered a necessity for practical nuclear aircraft. Accordingly the Lewis program is centered chiefly around these two areas. The accomplishments made in this study to date are presented in the following sections.

General Problem of Safety

1. The probable consequences of full fission-product release for flight over land indicate that fission-product containment is required for major aircraft accidents.

2. No fundamental reason has been found which eliminated the possibility of fission-product containment during impact of aircraft from a speed of 600 feet per second (183 m/sec). Techniques for absorbing the kinetic energy developed in such impacts have been envisioned and found to be fundamentally feasible. A great deal of experimental and analytical work is necessary to show that these techniques can be put into practice without excessive weight penalties.

3. No fundamental reason has been found that eliminates the possibility of fission-product containment after a major accident has destroyed all external afterheat cooling systems. The use of a containment vessel at least 10 feet (3 m) in diameter will provide enough surface area to permit removal of the reactor afterheat by thermal convection and radiation without melting and without introducing unreasonable weight penalties. Thorough design engineering and experimental studies are required to demonstrate that such containment is feasible in practice.

4. The probable consequences of full fission-product release for accidents occurring in overwater flight indicate that the severity of contamination is very much less than for overland flight. Fission-product containment may be required for political or psychological reasons.

General Problem Area of Long-Life

5. Molybdenum-water-moderated-reactor calculations have been verified by critical experiments. Calculations can now be made with confidence to determine power and fuel distributions for long-life reactors.

6. Burnups of 5 to 10 percent for molybdenum-UO_2 fuel elements have been experimentally demonstrated by a limited number of inpile tests. Reactor life of 5000 to 10 000 hours should be possible with such fuel materials. Further inpile experiments are required to fully establish burnup limits and operating lives of fuel materials.

7. Creep data required for long-life heat exchanger tubes have been obtained. A test of a heat exchanger header with tube-to-header joints has demonstrated performance capability of helium-to-air heat exchangers for a well-developed oxidiation-resistant high-temperature alloy at a temperature of 1550° F and a helium pressure of 1500 psi (103×10^5 N/m^2). Further work is required to demonstrate feasibility of representative heat exchanger assemblies or subsections in a simulated engine environment including flowing gases, thermal cycling, and vibration.

General Problem of Integrated Powerplant Concepts and Optimization Studies

8. First round studies of a liquid-metal fast-reactor system, a helium-cooled fast-reactor system, and a helium-cooled thermal-reactor system have been completed. The weight savings due to integration and multiple use of materials for shield and containment functions indicate the necessity for integrated powerplant design studies.

9. A powerplant-aircraft integration and optimization computer program has been developed.

10. Preliminary studies indicate that nuclear aircraft with gross weights of 1 million pounds (454 000 kg) designed for Mach 0.8 and 36 000 feet (11 000 m) should be able to carry a payload of 200 000 pounds (91 000 kg). Provisions for fission-product containment systems and reactor operating life greater than 1000 hours are included in these calculations.

Other Problems Areas

11. Existing or demonstrated liquid-metal technology for long life systems (up to 10 000 hr) particularly in the case of refractory-metal systems is lacking. The data that exist are sparse, conflicting, and inconsistent for systems that could produce turbine inlet temperatures much greater than 1100° F (866° K). Further detailed study of existing information may clarify the picture.

12. High-pressure inert-gas circulating system components require more study than has currently been given to assess the present state of the art. Experience with seals, turbomachinery, heat exchangers, and ducting for high-pressure inert gases is limited.

CONCLUDING REMARKS

In this attempt to assess the feasibility of practical, safe nuclear aircraft, no special missions are specified, except that they should be subsonic and require large payloads, long range, or long endurance.

The nuclear aircraft should be no more difficult to handle in normal flight and maintenance operations than conventional chemical aircraft. It must not produce any additional hazard to the flight crew, the ground crew, the passengers, or the general population than that for conventional, large, chemically powered aircraft.

These requirements demand (1) that complete shielding be provided around the reactor to reduce radiation dose levels to those acceptable by Federal Radiation Council standards, (2) that fission-product containment be provided for even the worst aircraft accidents if it is desired to fly over land without restriction, and (3) that even though the contamination problem is very much less severe for overwater flight, containment of fission products may be required for political or psychological reasons.

Technology is probably currently available to provide, within the above requirements, good performance for safe, practical, nuclear-powered aircraft with gross weights of about 1 million pounds (454 000 kg) or greater, providing the results of future fission-product-containment studies meet current predictions. Even though there are no fundamental reasons why fission products cannot be contained during and after major aircraft accidents, extensive analytical and experimental engineering studies are required to determine whether fission-product containment can be put into practice without excessive weight penalties.

Based on limited high-burnup fuel-element tests at nuclear aircraft reactor conditions, it is anticipated that reactor lifetimes approaching 10 000 hours may be feasible.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, August 3, 1967,
126-15-01-02-22.

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